Chapter 18

Radioactive Waste Management In Poland: Current Status of Investigations for Radioactive Waste Repository Areas

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Abstract: General information about the regulations and limitations concerning radioactive waste in Poland is given in the paper. Radwaste, 95% of which is low level at present, comes from one research reactor and over 2000 smaller producers. The responsibility for collecting, handling and disposal of all radwaste is delegated to one organization partially supported by the state. The status of new repository site investigations is reviewed.

18.1 Introduction

The suitable management of radioactive waste and spent fuel from research reactors appears to be one of the most important problems in Poland, because it has an impact on the safety and public acceptance of nuclear energy and the further development of this technology.

It is estimated that ionizing radiation sources in this country with an activity of about 200,000 Ci are being used in medicine, industry and scientific research activities. As of January 1, 1993, spent fuel from research reactors with an activity of 800,000 Ci (including 6000 Ci from transuranic elements) were in storage at the Institute of Atomic Energy (IEA) at Swierk. Decommissioning of research reactors is also a very important problem that should be taken into account in waste management programs.

18.2 Legislation for Radioactive Waste Management

Radioactive waste management in Poland is regulated within the framework of:

1. The Atomic Law, laid down April 10, 1986, by an Act of Parliament, in which the utilization of atomic energy for the public, as well as the economic needs of the country, is defined; and
2. A regulation from the President of the National Atomic Energy Agency (NAEA), announced May 19, 1989, on the Principles of Defining Waste as Radioactive, Classifying Them and Keeping Records, and the Immobilization, Storage and Disposal of Wastes.

To ensure safe transport of radioactive material, the IAEA regulations (Safety Series No. 6, 1985), and modal ADR, RID, IATA and IMO regulations are applied as appropriate. In practice, the transport of radioactive waste is only by road.

Radioactive waste is separated as follows: (1) beta and gamma emitters - high level (HLW); (2) alpha emitters; and (3) spent sealed radioactive sources. According to the regulation mentioned above from the NAEA's President, waste classifications are based on ALI principles as shown in Table 18.1.

18.3 Sources of Radioactive Waste in Poland

In Poland, radioactive waste comes from research reactors, scientific and educational institutions, industrial organizations and hospitals. Only low and intermediate level wastes are produced. The high activity gamma emitters in spent sources should be transported back to the supplier, but a number of them are still stored at different places in the country.

For many years, Poland has operated two research reactors and one critical assembly; and at present, a few thousand spent fuel elements from these reactors are stored at the site of the Institute of Atomic Energy. The storage facilities were originally planned only as tempo-

Table 18.1. Waste classifications.

Waste Form	Radiation	LLW	ILW	HLW
Solid	beta, gamma (ALI/m^3)	10^2-10^6	10^6-10^9	$>10^9$
	alpha (ALI/m^3)	$>10^2$	-	-
Liquid	beta, gamma (ALI/m^3)	10^{-2}-10^2	10^2-10^5	$>10^5$
	alpha (ALI/m^3)	$>10^{-2}$	-	-
Gaseous	beta, gamma (DAC)	0.1-10	10-10^6	$>10^6$
	alpha	>0.1	-	-

Note: ALI denotes a derived factor being the annual limitof radioactive intake through the alimentary canal (ALIp) or respiratory system (ALIo) for people employed in conditions of radiation exposure, stated in separate provisions.
DAC denotes a derived factor being the concentrations of radionuclides in the atmosphere for people employed in conditions of ionizing radiation, stated in separate provisions. DAC = ALIo/2400 m^3 where value of 0.1 DAC corresponds to the ventilation outlet.
In the case of unidentified isotopes, a more restrictive limit expressed in Bq/kg or Bq/m^3 may be used.

rary storage on the assumption that the spent fuel would be taken back by the Soviet supplier. The spent fuel is kept in a wet storage facility close to the reactor, and the age of the oldest irradiated fuel elements is 35 years. Conditions at the storage facilities are controlled by the user and by the National Inspectorate for Radiation and Nuclear Safety. The question of how and where this spent fuel is to be transported, stored or reprocessed appears to be one of the most important questions to be considered by the Government. Establishing a clear policy regarding the management of spent fuel seems to be one of the major elements having an impact on public acceptance of nuclear energy.

18.4 Organizations Responsible for Waste Management and Scope of Their Duties

According to the above mentioned Atomic Law:

- The issues falling within the scope of the Agency's activity is radioactive management;
- The head of the organizational unit within which the radioactive wastes arise is responsible for their handling in full conformance with the nuclear safety and radiation protection requirements and their preparation for transport and storage; and
- The head of the organizational unit that has been licensed to operate a radioactive waste disposal facility is responsible for keeping the radioactive waste in full conformance with the nuclear safety and radiation protection requirements.

The President of the National Atomic Energy Agency is designated, and can be recalled, by the Prime Minister and reports directly to him. The Management Committee, under the supervision of the President, acts within the Agency. The Committee adopts resolutions on matters related to the scope of the Agency's activities.

The responsibility for LLW/ILW over the entire country is delegated to the Institute of Atomic Energy. At present, practically all radwastes are collected, treated and conditioned at the IEA and disposed of at the Central Repository (CR) located at Rozan.

18.5 Treatment and Conditioning of LLW/ILW

The radioactive waste treatment and the conditioning methods at the IAE are aimed at reducing volumes and preparing for safe transportation and storage to fulfill the requirements for final disposal at the CR.

Low level liquid waste is chemically treated in a clarifier resulting in a volume reduction of about 100 times, and the sludge is transferred to a bitumization plant for further treatment. On the other hand, ILW is concentrated by evaporation, and the distillate is further purified by ion exchange before being released. The concentrates (evaporator sludge) are conditioned by cementation, and the radioactivity and chemistry of the decontaminated liquid effluents are controlled before being released. Solid LLW is compacted into 0.2 m^3 drums using a 12-ton press, and the biological wastes, after urea-formaldehyde conditioning, are stored in 0.05 m^3 drums.

18.6 STORAGE OF RADIOACTIVE WASTE

The Central Repository for radioactive waste is a near-surface type located 90 km from Warsaw on the grounds of a former military fort built in 1905. The CR was put in operation in 1961. The geology of the site is characterized by boulder and sandy clay. No historical records regarding seismic activities in the area are available.

Most of the repository is characterized by a concrete structure of military design with roof and wall thicknesses of 1.2 to 1.5 m, and a floor thickness of about 30 cm. Within a protection trench of the fort, a moat is used for final disposal with a concrete cover about 20 cm thick. Only solid and solidified low and intermediate level wastes are stored at the CR. After 33 years of operation, about 5400 m^3 of wastes have been disposed of in this repository. The cumulative activity of these wastes is 250,000 GBq (without decay), or 40,000 GBq with decay.

18.6.1 Storage in Concrete Bunkers

Concrete bunkers are used for temporary storage of alpha waste and contaminated installations and devices, which will be reused. Solid alpha wastes are placed in a chamber that is sealed off, after being filled, with a brick wall. Sealed sources of waste, with activities that do not exceed 4 GBq, are disposed of in one of the underground concrete bunkers. The access hole to this bunker is sealed with a lead cover lid 200 mm in thickness.

Except for the alpha waste categories, the LLW is disposed of in the moat of the CR where the bed and walls are made of concrete. The containers of the conditioned waste are placed in layers that are separated by layers of concrete. This procedure is repeated until the moat is filled to capacity, and the top layer is protected by asphalt.

18.6.2 Environmental Radiation Monitoring

The on- and off-site radiation monitoring system at the CR includes two basic groups of measurements:

- Radioactivity levels in environmental samples, and
- On-site and off-site gamma radiation levels.

Records of measurements are made by the IAE and presented annually to the National Inspectorate for Radiation and Nuclear Safety as well as to the appropriate local administration.

18.7 NEW REPOSITORY SITE INVESTIGATIONS

A study was initiated in Poland in the late seventies aimed at selecting areas suitable for radioactive waste repositories. Initially, the main attention was concentrated on selecting areas characterized by rock systems suitable for the permanent isolation of wastes. Salt beds, crystalline rocks and clay formations of considerable thickness were considered the most appropriate rocks for an underground repository.

The study was conducted upon the request, and was coordinated, by the National Atomic Energy Agency. Many specialists from various scientific institutions participated in the elaboration of specific issues. The investigations were designed to determine potentially useful repository sites in three categories: (1) superficial; (2) shallow underground; and (3) deep underground (see Nos. 1-10, Fig. 18.1).

18.7.1 Deep Underground Waste Repositories

An examination of geological formations that initially appeared suitable for the construction of deep waste repositories and could satisfy nuclear safety requirements led to the following selection of sites shown on Figure 18.1:

- Silurian shales in northern Poland (No. 5);
- Granites, bastard granites, crystalline Pre-Cambrian shales in eastern Poland (No. 6); and
- Triassic mudstone (No. 4).

The location of a repository within these formations

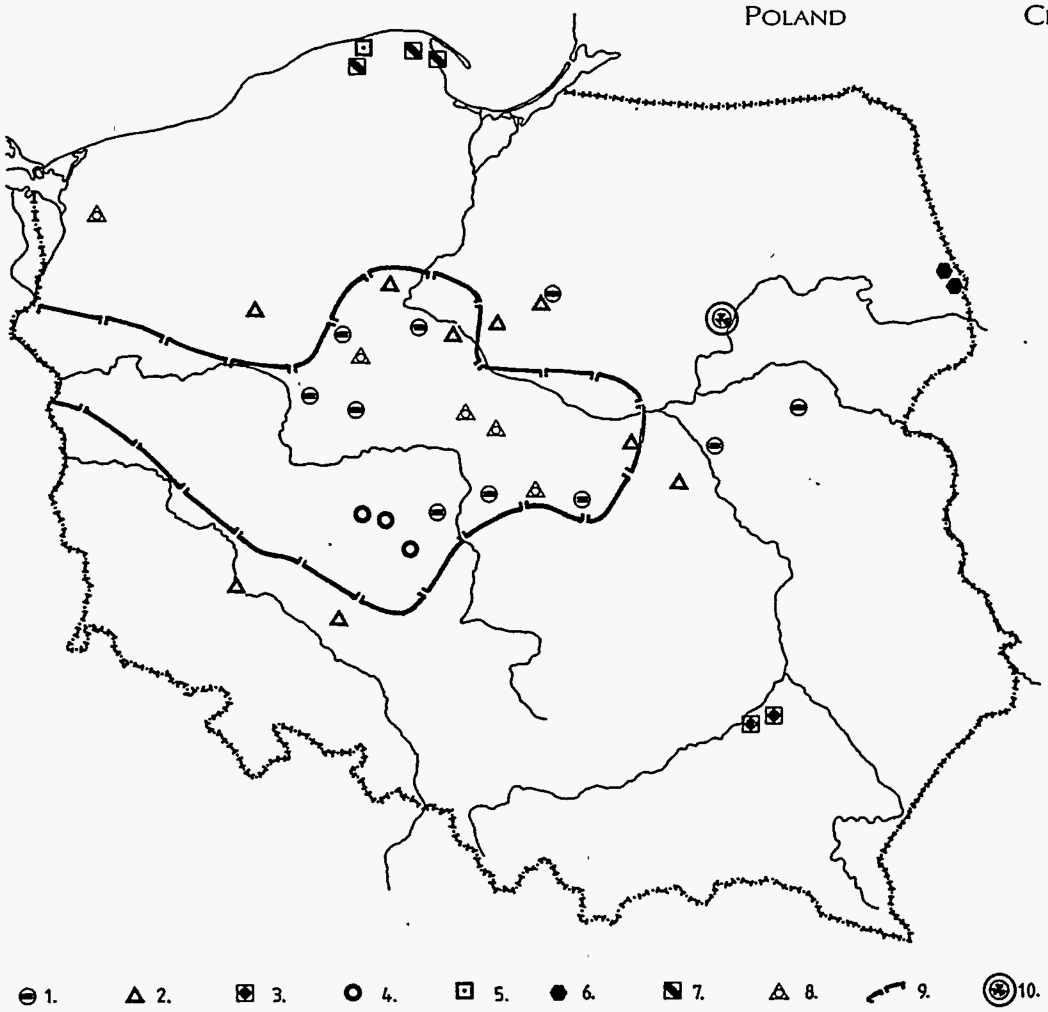

Figure 18.1. Areas selected in Poland for the possible location of a radioactive waste repository: 1 - superficial (Quaternary); 2 - shallow underground (Tertiary-Pliocene; 3 - shallow underground (Tertiary-Sarmatian); 4 - deep underground (Triassic)' 5 - deep underground (Silurian); 6 - deep underground (Pre-Cambrian); 7 - deep underground (Permian); 8 - deep underground(Permian); 9 - limits of area with favorable climatic conditions for superficial radioactive waste repositories; and 10 - active repository at Rozan.

would require the construction of deep underground works.

In discussions concerning waste storage in other rock types, the concept of storage in salt formations was also considered. The following selections were made from an analysis of formations of Permian and Miocene age:

- Rock salt in the Baltic region (No. 7); and
- Salt diapirs in central Poland (No. 8).

Information on rock salt has been revealed in varying degrees by wells and geophysical investigations. These earlier studies were conducted in terms of assessing the salt resources for the chemical industry. Solution mining (lixiviation) of storage cells in one of the rock-salt diapirs was considered to be the most appropriate solution.

During the period of planning for the development of nuclear energy, a study was carried out to determine the best waste repository policy to adopt concerning rock-salt beds in northern Poland. A method was presented for the prognosis of thermal effects in an underground repository for highly radioactive waste and the problems of optimizing the underground works and the storage technique. The repository was assumed to be located in Permian salt beds at a depth of 740 m beneath the surface. The average bed thickness in the area of the repository is about 200 m, and the overlying formations are anhydrides and Permian dolomite limestones. Above these beds are Mesozoic and Cenozoic formations with possible water-bearing strata.

To develop the prognosis of the temperature distribution, a method was proposed that is based on an analytical solution to the thermal conduction equation for the individual source (waste container). With regard to the design stages of the repository, this method allows one to more rapidly assess an optimal scheme for the distribution of repositories. The results of this model study showed that the thermal impact of such a repository becomes apparent only after some one hundred years and that the impact is practically negligible.

18.7.2 Shallow Underground Waste Repositories

Further considerations were therefore limited to the construction of a shallow repository in clay formations. The radioactive waste would be stored in shallow pits or large-diameter wells some 50-70 m under the surface. The following formations were selected:

- Krakowiec clays (Tertiary period - Sarmatian) in southeast Poland (No. 3); and
- Spotted clays (Tertiary-Pliocene) in central Poland (No. 2).

An analysis of the population conditions and physical management was carried out at these sites.

18.7.3 Superficial Waste Repositories

The existence of outwash sands that are located in boulder clays were selected as a favorable location for superficial waste repositories. The most favorable location on the watersheds of rivers is also essential. A study was carried out at ten locations using the methodology recommended by the International Atomic Energy Agency for the pre-selection and selection stages. The prospective areas were separated from major regions for further detailed analysis. An evaluation of the usefulness of the areas selected was carried out, in relation to an analysis of geographic conditions, on the basis of the existence of conditions that would exclude or limit the area itself. The characteristics of the areas were analyzed in terms of the following issues:

- Geological setting and hydrogeological conditions;
- Geodynamical processes;
- Potential of raw materials;
- Hydrology;
- Meterology and climate; and
- Environmental management and protection.

Most locations were concentrated in central Poland (No. 1 on Fig. 18.1) within an area where favorable climatic conditions for a superficial repository prevail. The following conditions were considered to be preclusive factors:

- Presence of legally protected areas (reservations, national and landscape parks);
- Areas with planned regional limitations on physical management;
- Areas under the influence of a concentrated groundwater exploitation;
- Areas with imminent 100- and 500-year floods;
- Areas where the subsurface waters are highly mineralized; and
- Areas with shallow groundwater.

An analysis of the social and economic conditions were also carried out in some of the locations. At the same time, attempts were made to obtain social acceptance for locations selected for a superficial repository.

18.8 SUMMARY

The nuclear energy development program in Poland is still not precisely defined, but it is clear that any further advancement in the field of nuclear technology cannot be pursued without solving the waste isolation problem. For the future of the national economy, there is a need for serious consideration of this source of energy. An increase in environmental protection studies is now quite evident and results in an enforced continuation of previous research concerning the the location and documentation of new repository sites.

The results presented in this paper are concerned primarily with the preselection stage. Only in the case of superficial radioactive waste repositories are some elements of the selection stage being investigated. Within a period of about ten years, the next studies will concentrate on the choice of possible locations for underground and superficial radioactive waste repositories. It is evident that, after potential sites have been selected, quite different programs of time-consuming research will be required for each of these types of repositories.

Chapter 19

Program Of Geological Disposal Of Spent Fuel And Radioactive Wastes In Slovak Republic

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19.1 Introduction

The present stage of spent fuel and high level waste management is a development from the nuclear cycle strategy of the former Czech and Slovak Federal Republic (CSFR). This concept was based on intergovernmental agreements between former CSFR and the Soviet union (USSR) on cooperation and assistance during construction and operation of Czechoslovak nuclear power plants (NPP). The cooperation program between CSFR and USSR provided for confirmed unpaid transport of spent fuel to USSR during the whole time period of operation of Czechoslovak NPP's. Up to 1988, about 700 assemblies of spent fuel were transported to the USSR.

Fulfillment of the above agreements was not fixed in the USSR approach. First, there was a prolonged period of spent fuel storage on CSFR territory, and after the political changes, new transport conditions, abandonment of unpaid transport, and a decrease of assembly pieces dedicated for transport. As a result, the Slovak Power Plants company decided to construct long period spent fuel storage. Wet storage is used in this operation and its capacity is 600 tHM (5,040 pieces of fuel assemblies).

The main reason for changing the original strategy of high level waste and spent fuel management was the abandonment of unpaid spent fuel transport after the political changes in the Russia Federation. With respect to the present economic situation in the Slovak Republic, it is impossible to decide on a new strategy in this field. Therefore, the Slovak Power Plants company decided to solve the problem with interim storage of spent fuel for approximately 50 years and start preparation of deep geological disposal of high level wastes and spent fuel under the conditions of the Slovak Republic.

19.2 History of project development

The history of repository development designed to accept high level and long lived radioactive wastes and spent fuel started two decades ago. At that time the Nuclear Research Institute Rez (Czech Republic) issued some basic policies concerning the conceptual and safety aspects of a deep geological repository.

The year 1984 became a turning point for the intensification of activities towards underground facilities. Initiated by Energoproject with participation of Gasproject, and Construction Geology, the Nuclear Research Institute and other companies examined the possibility of constructing a deep silo (over 500 m) for reactor waste from NPP Dukovany and later NPP Temelin. Even though the project was abandoned, it provided valuable experience from the fields of geology, safety studies and contacts with the public.

Further activities, stimulated mainly by the need for processing waste arising from the decommissioning of NPP A-1 in Jaslovske Bohunice, were more precisely defined. They sought tools usable for evaluating the acceptability of sites, the solicitation of proper regions on the basis of archival data, and the extension of possibilities to provide safety analysis. The leading companies performing these activities were Geoindustry, Central Institute of Geology, Institute of Geophysics, Nuclear Research Institute, Nuclear Power Plant Research Institute, Dionyz Stur Geological Institute and others. One of the achievements reached was a basic evaluation of the Slovak territory for purposes of siting a deep geological repository.

It is reasonable that works aimed at the geological repository were performed by a number of institutions, and their costs were covered from different sources. This led to a certain heterogeneity of activities and as a consequence an incongruity and incoherence in the results obtained. Therefore, urged by a need to initiate a program of spent fuel management, which arose after the political changes in Europe, the Federal Ministry of Economy of CSFR started the preparation of a technical program initiating the most urgent works in 1992. The purpose was to concentrate available agencies in a consistent, centrally coordinated contract. In the course of preparing this task, practically all central organizations involving contractors, as well as most research bodies active in waste management as suppliers, were involved. Firstly, the technical content of the project was defined and then, as a result of competition, the Nuclear Research Institute was appointed as coordinator. Unfortunately, due to the division of Czechoslovakia, funding of the project collapsed. To promote the primary purpose of the project, in the spring of 1993, the Czech Power Company and the Slovak Power Company decided to order a study on, "The plan for development of a deep geological repository" and to share its expenses equally.

In view of the importance of the report for further progress in repository development in the future, it was decided that the contractor should ensure an international review of the prepared document. For this purpose, a contact was established with the administration of The Nuclear Cycle Division of the International Atomic Energy Agency, which advised that the government ask for an evaluation within the Waste Management Assessment and Technical Review Program (WATRP). The results of the mission were attached to this study as a separate document in December 1993.

After the division of CSFR, work on development of deep disposal of high level wastes and spent fuel in Slovak Republic started only at the end of 1995. Because of the developments since 1993 in world-wide experience with deep disposal, and because the original project was elaborated for conditions in the former CSFR, the first step in continuing the work was the need to revise the original project for deep disposal development and adapt it to conditions in Slovakia.

19.3 PURPOSE OF PROJECT

The main goal of the revision was to specify the main bounds among the particular tasks of deep disposal development. This means taking into account all requirements for site selection, near-field and far-field interactions, quality assurance, safety analyses, the role of the public as well as design studies and licensing steps under the conditions of the Slovak Republic. The requirement of the customer was to compile a basic overview of activities aimed at the construction of a deep repository designed for spent fuel and for long lived and high level wastes. Especially, the research, development and design phases were to be thoroughly elaborated so that connections within each topic as well as between different problems are respected.

The technical characteristics of solutions for each topic had to be completed with consideration for time and economy; however, there are some doubts about the reliability of both tasks. The reason is simple. Development of the deep repository needs a very long time (tens of years), and thus, it is possible to make only extremely rough estimates of the time demands, and as a consequence, the economical needs of particular tasks. Furthermore, the time consumption is often dependent on non-technical issues, such as: the influence of public acceptability of particular solutions, licensing period, changes in development procedures due to political decisions, probability of receiving unacceptable results following the repetition of a certain volume of work, consequences of changes in legislation, etc.

The next requirement of the customer was to postulate in detail a 5-year program of work respecting the programs in progress, underlining within each topic the main activities that could, when postponed, retard solution of other development problems, and initiating new key tasks that have not yet been started.

The document, which is the result of the deep disposal project revision, will serve as a guide for all necessary research and development procedures that must be coordinated, and completely and equably answered in all aspects.

19.4 RESULTS OF PROJECT REVISION

To reach all claims during a very short period, the following procedure has been adopted:

- a philosophy of the study was selected that consists in parallel solutions of particular tasks by working groups supervised by an institution and a specialist experienced in waste disposal;
- some comprehensive problems were defined and

supervisors for each of them were nominated;

- nearly every three weeks, coordination meetings were organized in which the progress and results obtained were evaluated and critiqued, and goals for the next period were set up; and
- the final versions of the respective parts were compiled in this document and completed by introductory chapters and annexes.

The above mentioned procedure resulted in a structure of the document which describes a gradual improvement in each problem in connection with new developments in the field of deep disposal as well as their adaption to conditions in the Slovak republic. It is clear mainly when inspecting the diagrams. The diagrams are complemented by commentary that should explain the content of each of the activities. Its other role is to mention problems that could not be easily read from a diagram. The description of each particular task contains discussion about the economical and time aspects for the solution of the topic.

The detailed 5-year plan of activities describes the solution for each particular task in the development of spent fuel and deep disposal of high level waste until the year 2000. The plan contains concrete data on time and economy that are necessary for the deep disposal project realization. The main aim of the short-term plan is to show waste producers, as well as contractors of revisions, an extent, a content and a cost of anticipated works for the preparation of the deep repository construction. In this way, the plan can serve as a background document for planning purposes.

The basic result of the project revision, in which the solution of the whole project is shown, is a "Diagram of deep repository development" (Fig.19.1). This diagram is vertically divided into the main particular tasks, following the text of this report. Horizontal divisions indicate a time succession of activities in each topic and, to a certain level, it also parallels the implementation of the main works. It should be mentioned that the diagram is significantly simplified and that the time axis is not linear, which means that there is no correspondence with real timing of the topics being considered .

The diagram shows one interesting point. Practically every particular task, such as research and development, consists of several phases. They include construction, operation and closure of a facility, and they even include some data on the course of a final evaluation of safety and reliability of the system. When considering that this concerns a time period of more than one hundred years, then the role of a coordinator of all activities, credible working groups, and uncompromising control, supervision and licensing bodies must be stressed.

The main branch of the diagram is connected to a topic "Design activities and construction". This line comprises all principal decision processes. It also includes the key outcomes, design studies and designs, and construction procedures. It can be stated that the other main particular tasks provide the necessary data for decisions of a designer concerning the final solution of an underground repository and auxiliary facilities.

Geological works are connected to an evaluation of the mobility of contaminants in the geological system (far field interactions). Activities contained in this double-topic are focused on selection and verification of the suitability of a site and corresponding geological system for construction of a repository. They also summarize input data for safety analyses of long term behavior of the system.

Studies of the behavior of engineered barriers, final waste forms, packages, overpacks, sealing and filling materials, and underground constructions (near field interactions) are aimed at a choice of an optimum composition of barriers and to evaluate the effectiveness of their retention ability for radiocontaminants. Other species of disposed materials are to be evaluated as well (source term). The results of these studies provide proposals for material composition of repository construction addressed to designers, and data dealing with velocity, mechanisms, and probability of release of immobilized species.

The task of safety analyses is to summarize and evaluate all data received in the course of geological investigation and research, during construction and operation of a facility, at waste treatment processes. In addition, they lay down limits and conditions that any construction structure shall fulfill. Elaboration of safety analysis is the necessary condition for any licensing procedure, and thus, it can be considered to be the absolute key topic of repository achievement.

Quality assurance belongs to a group of activities designed to secure the maximum level of biosphere preservation. Its aim is to work out programs of control and supervision of all activities connected with reposi-

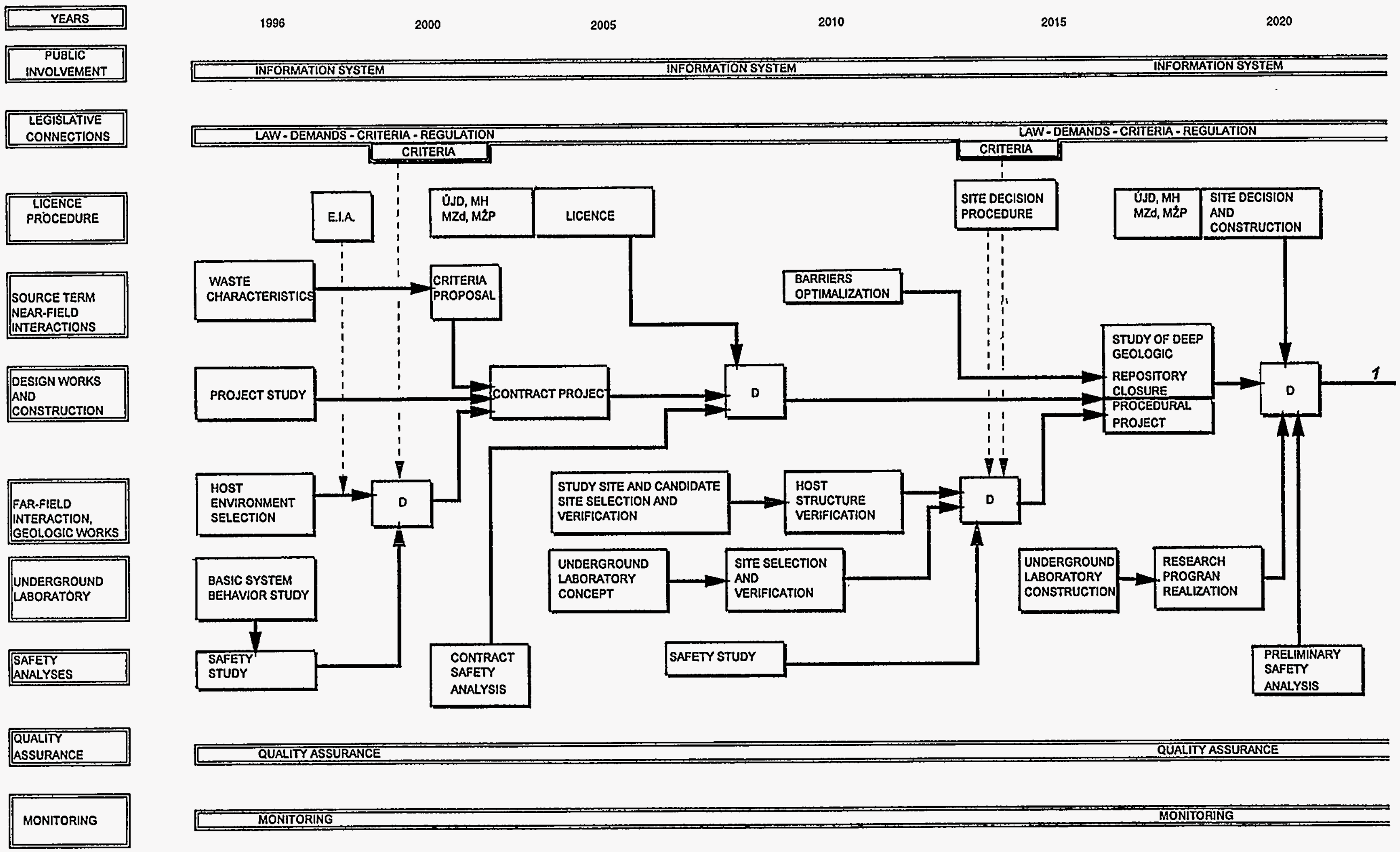

Figure 19.1a. Diagram of repository development.

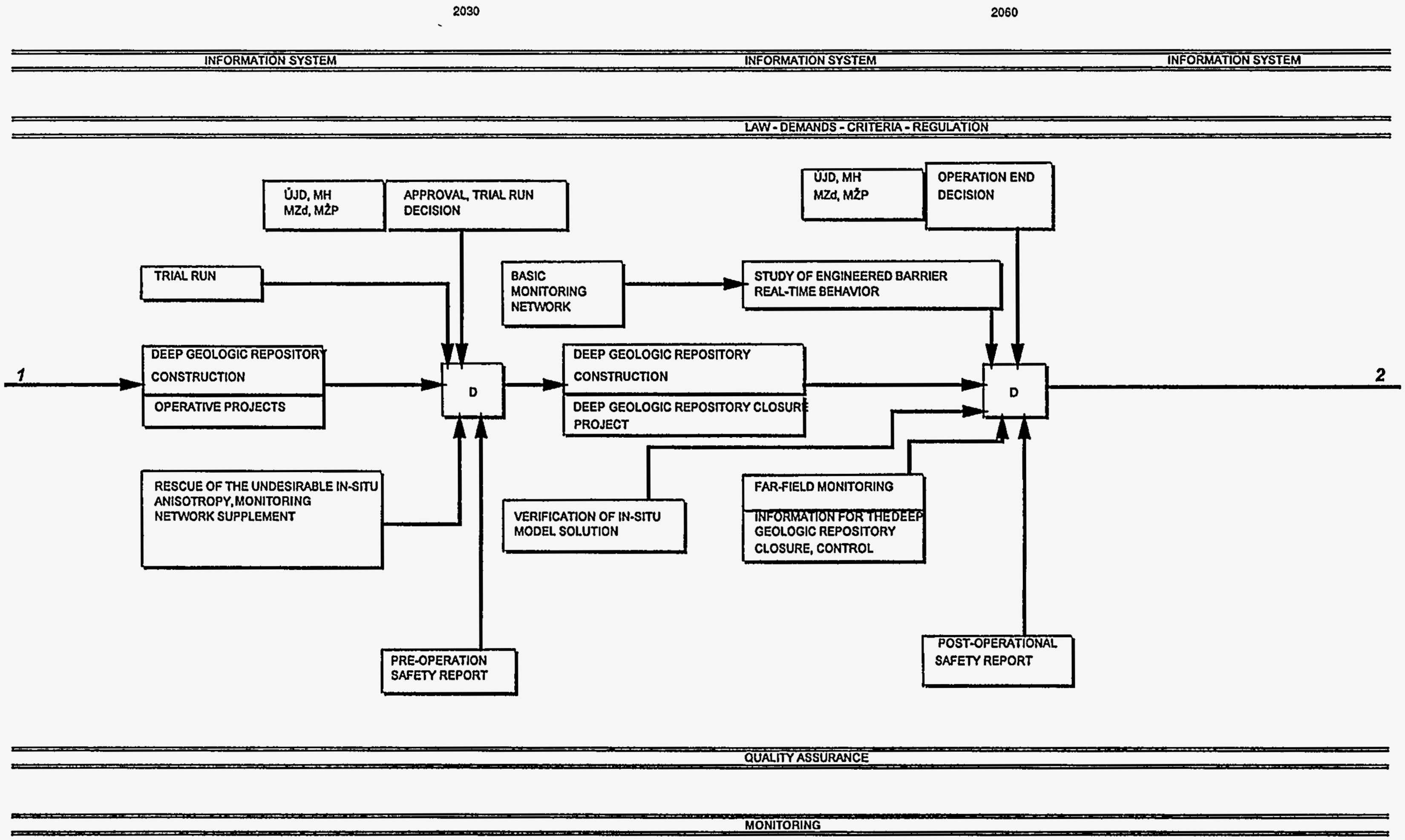

Figure 19.1b. Diagram of repository development (continued).

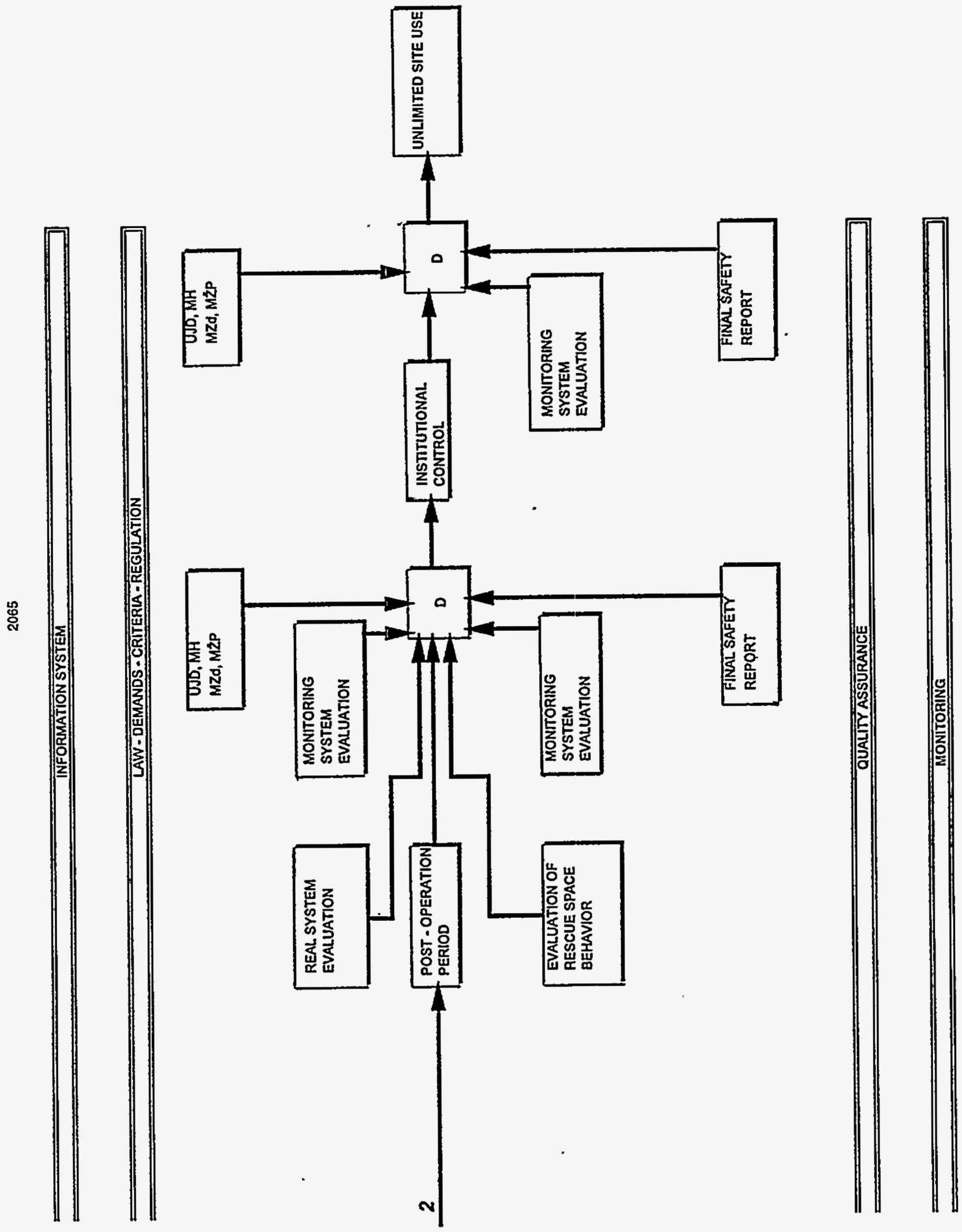

Figure 19.1c. Diagram of repository development (continued).

tory development and construction. The goal of these procedures is to eliminate risks mainly connected with the so called "human factor".

The licensing processes are included in the particular task "legislation connections" because these processes influence significantly all other steps of deep disposal development. They permit each subsequent activity; they may on the basis of some independent evaluation prevent the implementation of such steps. Actually, they are connected to the formulation of claims, requirements, and methods. Sometimes they even set the conditions, and the licensing decisions are issued using an interpretation of these limits and recommendations for a concrete system.

Monitoring in the diagram (Fig.19.1) is considered to be an independent, permanently valid particular task, although in the study, it is described in other particular tasks. Deep disposal monitoring includes a number of steps: radiochemical, hydrological and hydrogeological monitoring of sites from the beginning of research; geological monitoring of a rock structure; measuring the radiological impacts to personnel and the public; recording the changes of state and behavior of barrier materials; meteorological records; observation of destructive and corrosion phenomena, etc. Interpretation of the monitored data is one of the basic conditions required for closure and release of the facility.

The results of particular tasks of the public may by its consequences not only influence but completely change any technical decision. Public involvement in the process of developing a deep repository has at least two aspects. First, passive, which is interpreted as an information campaign about the repository system, design, construction, safety, risks and advantages of its realization. Second, active, which is an effect of public meaning and also the design of the repository by independent opinions and evaluations. A first-rate program on communication with the public may simplify the achievement of the repository; on the contrary ignoring the necessity of reaching a concensus with the general public can completely eliminate the project.

The revised document for a project of deep disposal development is seen first of all as a methodological document. Particular problems of the project may differ in their content, but the approach to their solution has some common features, such as long-term considerations and the principle of conservatism.

The general feature of systems of radioactive waste management is the long-term consideration of all processes, activities and applicable phenomena. This is shown by the fact that partial issues are converted into material outputs after a long period, often reaching tens of years. Time factors bring a number of questions to an evaluation of the behavior of repository elements. To answer them requires finding unusual and substitute ways, e.g. mathematical modeling, studies of natural or man-made analogues.

An indirect interpretation used for waste form behavior during disposal is rather lengthy, and it involves some uncertainty. To reach a desired level of safety and functionality of the system, a principle of conservatism is applied in any evaluations. This means that those phenomena, or some combination, are considered that bring less favorable results.

The important fact in the methodology of deep disposal development is to have an objective approach to any step, activity or decision. That kind of solution is provided by preparation and realization of quality assurance programs. The main part of those programs is multiple opinions and evaluations of solutions. There is a tendency to leave out this demand. This is dangerous from two points of view. Any incorrect decision may cause irreparable damages even resulting in canceling the previous results, or the process of repository development may be suspended by a qualified opposition because of the inability technically to defend chosen solutions.

19.5 CONCLUSION

The revision of this project is the result of the work of a group of employees of the following institutions: DECOM Slovakia Ltd., Nuclear Power Plant Research Institute, Geologic office of Slovak Republic and EPG Invest Ltd. It has been worked out under the technical and editorial coordination of the DECOM Slovakia Ltd., however, separate authors are responsible for their own sections. All of those involved have attempted to develop an approach in the most objective way.

The goal of the activities was to perform a revision of the project from the year 1993, in which must be included the deep disposal development of the former Czechoslovakia, to incorporate the new world results of this field development, to modify it for conditions of Slovakia, and to add the economic and time considerations to the technical solution. The determination of concrete activities through the year 2000 was an inseparable part of this methodological document. Each particular task of the project contains ideas about concrete activities, financial costs and solutions as well.

A summary of the above mentioned data in one document creates the material, which defines concrete activities and the needs of finance to assure the required output from a solution for a deep repository development under the conditions of Slovakia.

CHAPTER 20

GEOLOGICAL ASPECTS OF SITE SELECTION FOR LOW AND INTERMEDIATE LEVEL RADWASTE REPOSITORY IN SLOVENIA

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Abstract. According to the guidelines for a low and intermediate level radwaste repository site selection in Slovenia, the siting process has been divided into four steps. The first three steps of the surface site selection were completed in 1993. A set of the most exclusionary geological criteria to be applied in selecting the surface site is described. Some reasons for the failure of this process are also described. Since the fourth step was stopped due to strong public opposition, an alternative of underground disposal is now being considered. In 1994, the Agency for Radwaste Management started the preparation of basic guidelines for underground repository site selection. Joint recommendations, that consider both surface and underground site selection parameters, are now being developed in the Slovenian Nuclear Safety Administration.

20.1 INTRODUCTION

The guidelines for the selection of a low and intermediate level radwaste (LILW) disposal site were setup in 1991. The guidelines that were announced included rules according to which, under the given urban and social conditions, the most suitable site for the shallow ground disposal of LILW in Slovenia would be selected. Both the existing world-wide experience and the national regulatory conditions in Slovenia were considered in creating these guidelines.

In selecting disposal sites, it was necessary to have a detailed knowledge of the process of migration of contaminants into the biosphere. Slovenia has a very sophisticated geological and tectonic setting dominated by various combinations of geological structural elements such as: faults of different type and age, overthrusts, folds, naps and lateral transformations of different lithologic units. In most cases, it was very difficult to determine the migration of radionuclides in underground water. Thick layers of impermeable rocks are the only reliable natural barrier in such geological and hydrogeological conditions.

However, the requirements given by the guidelines are that a shallow disposal site is to have rocks of low permeability in the basement, and a distance to the underground water table that is as large as possible. Sites with these geological conditions, such as saturated clay marls, were the only ones selected as being acceptable. These rocks, regardless of fracturing in neighboring layers, provide a sufficient natural barrier to prevent migration of radionuclides.

20.2 SITE SELECTION PROCESS FOR LILW

The procedure used in selecting disposal sites was divided into three steps containing 43 criteria. In a final fourth step, the technical confirmation was based on a detailed field examination of the geology, hydrogeology, and seismology of the site. Each step was terminated by a presentation to the public of the results.

In the first step, unsuitable areas were excluded by taking into consideration certain exclusion criteria, such as: national parks, urban zones, ground water resources, presence and location of active faults, geothermal areas, flood areas, presence of ores, minerals, oil, gas, hydraulic conductivity, soil composition, thickness and extent of geologic units.

In the second step, the remaining acceptable areas were evaluated according to land use, water resources, seismic and geological criteria, so they could be further reduced to so called potential sites.

In the third step, several of the most suitable of the potential sites were chosen by comparing their locations on the basis of the following criteria: population, economic feasibility, transport, ecological value, and public acceptance.

In the final fourth step, a comprehensive analysis of the most suitable sites from the third step was carried out by applying the criteria of the previous steps and additional criteria concerning the corrosion of waste containers (biological processes, chemical properties of the soil and groundwater), and then a detailed field investigation was carried out to confirm the suitability of the sites. The results of the fourth step produced one or two of the most suitable sites that were considered to be technically confirmed. A schematic diagram of this process is shown in Figure 20.1

20.2.1 Step One of Site Selection

In carrying out Step One, a series of overlaying maps were used which contained areas that are defined by seven exclusionary criteria as described in Table 20.1. This process eliminated the unsuitable areas of the Republic of Slovenia from further consideration.

After considering the exclusionary criteria of the first step, the acceptable areas for an LILW repository site in

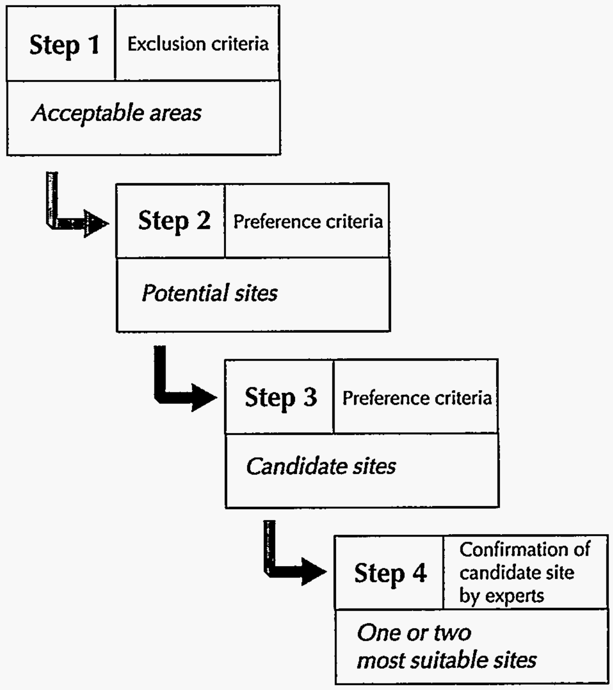

Figure 20.1. Schematic diagram of the site selection process for LILW in Republic of Slovenia.

Table 20.1. Exclusionary criteria of Step One.

Exclusion Criteria	Explanation
National Parks	The areas defined as national parks are excluded.
Urban Zones and Settlements	Excluded are all areas defined as settlements with more than 5000 inhabitants.
Drinking Water Resources–Aquifers	Excluded are all areas defined as drinking water resources.
Known Active Geological Faults, Geothermal Areas and Seismicity	Excluded are all areas located on a known active fault at a distance up to 3 km and the areas where the expected earthquake acceleration exceeds 0.3 g.
Flood Areas	Excluded are areas which are located in an area of 500 year floods.
Presence of Ores, Minerals, Oil and Gas	Excluded are areas with proven resources of ores, mineral, oil and gas.
Geological and Lithological Soil Composition	Excluded are the areas where surface homogeneity of layers is smaller than 300x300 m and the quotient between the thickness and hydraulic conductivity of layers is smaller than 5×10^9 s. Excluded are lithological layers having a hydraulic conductivity greater than 1×10^{-8} ms^{-1} and a thickness of layers smaller than 20 m.

Slovenia were identified. The potentially acceptable areas were those that had not been excluded according to any criterion of Step One. All of these areas were considered to be equivalent, i.e., the acceptable areas had not been assessed and evaluated. Figure 20.2 shows the locations of the acceptable areas after the application of the first step.

20.2.2 Step Two of Site Selection

In carrying out Step Two, the preference criteria were divided into four groups: geological, seismic, land use, and potential water management. These criteria were then applied to the acceptable areas selected in the first step.

The following geological preference criteria were applied:

- Presence of groundwater;
- Site seismicity;
- Presence and vicinity of active faults;
- Exploitation of ores/minerals, oil and gas;
- Areal extent of host rock;
- Thickness of rock mass;
- Soil instability;
- Erodibility;
- Rock composition and hydraulic conductivity;
- Angle of slopes; and
- Radionuclide paths to the biosphere.

The result was the selection of 36 potential sites occupying a total area of approximately nine km^2.

The examination of the potential locations was performed at the end of the theoretical studies to verify the procedure, and to determine discrepencies in the results obtained. This examination resulted in an expert conclusion that: (a) five locations are not suitable for the construction of a repository, and (b) another five locations are only suitable for a tunnel type repository and not for a surface type as previously envisioned. One potential site, suitable for both types of repository, was also identified and considered in further analysis.

The results of the second step of surface repository site selection were reviewed by a group of experts that confirmed the accordance of the procedure with the guidelines.

20.2.3 Step Three of Site Selection.

In the the third step, five candidate sites were selected among 36 potential sites from the second step. The

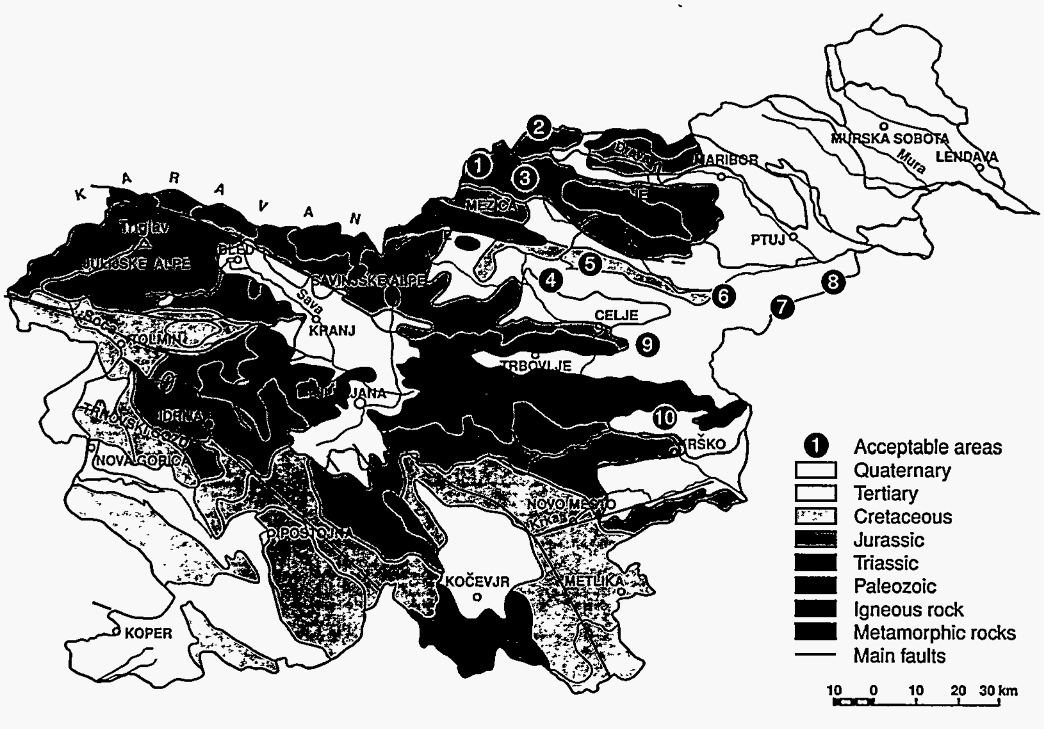

Figure 20.2. A generalized geological map of Slovenia with acceptable areas after the application of Step One.

method of assessment was based on the repeated use of the criteria from the first two steps. In addition, preference criteria concerned with economic and technical feasibility, transport, and social acceptability were considered as well.

One of the selected sites was found to be suitable for a surface repository, two, for a tunnel repository, and the remaining two sites were appropriate for either type of repository, surface or tunnel. The main geological characteristics of the sites are summarized in Table 20.2.

In accordance with the practice from previous steps in this procedure, the results were presented to the public. The presentation was not successful and has provoked strong disapproval within the local communities. Their representatives declared that waste disposal in the vicinity of their communities was not acceptable. It was evident that public acceptance of the candidate locations could not be achieved. Therefore, it was impossible to proceed to the fourth step in which the most suitable locations could be verified and approved by the experts. The project was stopped.

20.3 DID APPLICATION OF GEOLOGICAL CRITERIA INFLUENCE AN UNSUCCESSFUL SURFACE REPOSITORY SITE SELECTION?

It is clear the natural site characteristics play an important role in the selection process for a radioactive waste

Table 20.2. The main gelogical characteristics of the sites.

	Main Geological Characteristics	Site 1 Surface Repository	Site 2 Tunnel Repository	Site 3 Tunnel Repository	Site 4 Surface or Tunnel Repository	Site 5 Surface or Tunnel Repository
HOST ROCK	Rock Type	Sandy marl	Sandy marl	Sandy marl	Marl	Marl
	Permeability (m/s)	10^{-9}-10^{-11}	10^{-9}-10^{-11}	10^{-9}-10^{-11}	$10^{-11} < K < 10^{-9}$	$10^{-11} < K < 10^{-9}$
	Relative Porosity (%)	22	22	22	20-33	20-33
	Thickness of Layer (m)	300-400	300-400	300-400	50-400	50-400
	Areal extent of Rock Mass (ha)	11	332	104	19	28
	Angle of Slope (°)	10-20	5-20	5-20	10-15	10-20
	Erodibility (mm/300 yrs.)	4.2-5.4	3.2-7.5	6.4-8.6	5.4-6.4	4.2
	Point Load Index I_s (50) MN/m^2	0.87-1.94	0.87-1.94	0.87-1.94	0.09-0.31	0.09-0.31
	Unconfined Compressive Strength (MPa) $Q_u=22\ I_s$	17.17-42.71	17.17-42.71	17.17-42.71	1.99-6.73	1.99-6.73
	Natural Volume Weight (KN/m^3)	20.95	20.95	20.95	18.22-19.43	18.22-19.43
	Distance from Active Faults (km)	3-4	3-4	3-4	3-4.6	3-4.6
	Max. Expected Horizontal Acceleration in a Time Period of 1000 years (cm/s^2)	190	190	190	250	250
	Max. Expected Local Intensivity in a Time Period of 1000 years (MCS)	8	7-8	7-8	8	8

repository site, and that a site within an appropriate geological environment is, to a great extent, based on geological conditions.

But geological criteria, applied to the process of selecting a surface repository site in Slovenia, used only properties of the geological barrier and took no account of the other two barriers, i.e., conditioned waste and engineered barriers. In other words, the objective of finding a site for a surface repository was *to find a location with geological properties (natural geological barriers), where engineered barriers would not necessarily be used to achieve the safety standards.* This was certainly the most economic way for repository construction, but on the other hand, the site selection was exceptionally difficult.

The siting process applied highly quantitative exclusionary or preference geological criteria, i.e., geological criteria were stated with numerical values of the geological parameters, that made the whole siting process very inflexible. Some geological criteria, according to their "importance" (according to the size of the excluded areas in surface repository site selection) are presented and described in the following.

The criterion of "active faults" as a single tectonic exclusionary criterion has eliminated 97% of Slovenia for the purpose of siting a repository. To meet this criterion, areas located in the vicinity of a known active fault at a distance up to three km were unsuitable.

Considering the same criterion in the second step, i.e., presence and vicinity of active faults, acceptable areas from the first step ranged over the following distances:

- Unsuitable sites, where the site was to be located on, or near a fault, at a distance up to 3 km;
- Less suitable sites, where the distance from the fault is 3 to 8 km; and
- Suitable sites, where the distance is greater than 8 km.

It should be noted that up to the third step, the work included office work only, and no site investigations were performed to confirm activity of the faults.

Although Slovenia lies in a seismic territory, and tectonic causes of seismic activity, i.e. surface faults, are distributed all over Slovenia, there is a basic question (that could be discussed) whether the application of a uniform step-off distance is a matter of policy rather than being grounded on technical principles. Without detailed site investigations, it is difficult to select suitable locations, and the geological properties of specific sites must first be confirmed through field investigations.

According to the criterion for "Active faults", the WAMAP mission[5] recommended that at an early stage, it is important to decide on the definition for an active fault and the significance that rock structure could have on the integrity of a repository over its 300-year assessment period. It is necessary to have a single representative data base, or set of maps, supporting the interpretation and application of this criterion.

A similar situation occurred in the first step in connection with the "Lithology" criterion, where rocks with a hydraulic conductivity greater than 10^{-8} m/s, a thickness of layers less than 20 m, and a seismicity where earthquake accelerations greater than 0.3 g would be expected, were recognized as unsuitable. In further analysis, the Lithology of acceptable areas was compared considering the preference criteria "Areal extent of host rock" and "Thickness of rock mass", where the area suitability increased with extent (greater than 300 x 300 m, i.e., 9 ha) and layer thickness (more than 20 m). Again, this site analysis only involved office work. No field investigations, to confirm exclusive parameters for the rocks, was made.

In considering the preference criteria "Site seismicity", acceptable areas ranged between unsuitable (where the expected earthquake accelerations a_{max} exceeds 0.3 g for a period of 1,000 years), less suitable (a_{max} = 0.15 to 0.3 g) and suitable (a_{max} is less than 0.15 g). The maximum horizontal ground acceleration for the territory of Slovenia was evaluated with a probabilistic seismic hazard analysis.

The mission report[5] suggested that in general, an application of highly quantitative exclusion or site preference criteria, especially at an early stage of the selection process, was not recommended.

Quantitative criteria, as applied in the siting process and as described above, can only be used where quantitative data are available to justify their use, i.e., data confirmed by site investigations. Much of existing technical data is regional (non-site specific) and qualitative in nature as well. Some criteria, used in the first (exclusionary) step simply assume certain site specific data, which would only be available in the necessary detail after a careful site investigation. Such criteria should therefore be left to the appropriate later steps in repository siting[5].

It is very important to recognize the uncertainties in understanding the geology during the siting process, when data are based only on office studies. The actual site conditions may be significantly different from those envisaged, and as a result, the site selection process must remain flexible enough in order to accomodate unexpected features. More confidence can be placed in sites selected in a location where the geological structure is non- or less complex.

20.4 NEW APPROACHES

According to the fact that the necessity for the final disposal of low and intermediate level radioactive wastes is growing, the final location of the disposal site should be selected within the next five to ten years. The existing wastes are temporarily stored in interim storage facilities located at the Krsko nuclear power plant.

It is obvious that problems concerning final disposal of LILW should be solved in a satisfactory manner in the near future. Solutions for this problem are being searched for in the following directions:

1. In verification, new estimates and corrections of the three most exclusionary geological criteria (active faults, seismicity, and hydrogeological parameters), but the most important features have been revealed in the application to sites for final waste disposal.
2. In considering the newest techniques and technologies that have been developed in disposing of, and protecting, radwastes in the developed countries, and in reconsidering geological criteria in this new light.
3. In taking into consideration the possibility of underground waste disposal of LILW in geological structures, and in this way minimizing the risks arising from the seismicity and activity of fault zones.

In the analysis of the reasons for failure in the first campaign, it appeared that there was a bad coordination between experts in the different fields of science. For example, geologists considered "impervious" rock as the only suitable rock for a disposal site, regardless of the possibility of using engineered barriers (such as, canisters, filling materials, etc.). It is well known that over a period of 300 years, which is the time necessary for the radioactivity to decay to normal levels, it is possible to produce effective engineered barriers. Therefore, the requirement for an "impermeable" basement beneath the deposit is no longer necessary. On the contrary, in some repositories, such as Centre de l'Aube, a permeable basement is part of the design of the facility.

The Agency for radioactive waste disposal in Slovenia has also noted this deficiency from the first campaign. The new approach was therefore to provide some basic technology to the experts who don't have much experience in dealing with problems of packing and deposition. In this way, the Agency expected to ameliorate the cooperation of these experts with that of others.

In accordance with this new policy, the Agency has redirected geological experts to review the new technologies in the field of radwaste disposal in the developed countries. A series of such reviews have been carried out in which the first aim has been achieved; the geological experts of today are well acquainted with the technological possibilities and requests for construction of surface, or underground, disposal facilities. In addition, the Agency has made it possible for some of the experts to visit existing sites, and to meet other geologists and experts in other fields of science at international conferences. In this way, our geologists not only gathered new data, but also established contacts with colleagues from different European countries, exchanged opinions and learned new ways of thinking. It was especially useful for us to learn of unpublished experiences (both good and bad) that led to the solution of problems on multinational projects (such as the underground laboratories at Mol in Belgium, Grimsel in Switzerland, etc.).

Based on this new knowledge, a set of six possible types of disposal facilities has been defined for Slovenia, which include geological and rough technological conditions. They provide a basis for new considerations and estimations in carrying out campaigns of field investigations.

The existing criteria from the first campaign have been thoroughly reexamined. The result is a new approach in the evaluation of the exclusionary criteria. The philosophy has changed; the elimination of a site or a region on the basis of a certain criterion should be based on direct or indirect evidence.

There is another novelty in our way of thinking; we no longer look for the geologically best location, but for all acceptable locations. In this way, these locations are also available for analysis using other necessary criteria. We no longer have criteria for site selections or the elimination of territories, but guidelines that can give us an indication of possible problems. This approach does not limit our decisions in advance, and thus enables a more flexible treatment of the site selection process.

Since the site selection process for a surface repository has been stopped, this new approach is more likely to gain public acceptance by disposing of radioactive wastes in an underground facility.

The expansion of the site selection program to include an underground disposal facility gives us another possibility, and is the result of the new approach over the last few years. New geological guidelines for underground low and intermediate level waste disposal have been made and revised, and on this basis, new geological guidelines for surface disposal of LILW have also been remade.

Since some geological criteria are more important and can be more applicable to underground than to surface repository site selection (or vice versa), the proposed criteria differ, in many respects, from those for surface site selection. With regard to seismicity for example, underground structures are less susceptible to seismic disturbances than surface structures due to the fact that effects from earthquakes diminish with depth. Different transport pathways for radionuclide migration through groundwater to the biosphere should be considered in both site selection processes as well.

By placing the disposal system underground in rock means, on the one hand, having the possibility to minimize the influence of the most selective criteria used for a surface repository; and on the other hand, providing an underground disposal facility that, hopefully, would be more acceptable to the public.

The new proposed guidelines for underground LILW disposal consist of the following main parameters. (We are presenting them here to show the differences with the first criteria used in the selection process for a surface site.)

- Geological rock structure
  - volume
  - simplicity
- Lithology
- Hydrogeological conditions
  - permeability
  - hydraulic gradient
- Migration
  - geochemical properties of rock and soil
  - geochemical properties of groundwater
- Active endogenetic processes
  - seismicity
  - recent fault movements
  - volcanoes
- Rock disturbance
  - human reasons
  - natural reasons
- Potential resources
  - value
  - genesis
  - technology
- Geomorphologic stability
  - surface stability
  - water degradation processes
  - extreme climates
- Geomechanical conditions

The Agency for radwaste disposal, being responsible for the site selection process in Slovenia, will have to use this new approach and also help it to find its way to the public. Reports of all studies made are available in the Central Technical Library, and summaries of these studies are translated into English. This enables all concerned to be kept informed about the dangers, scientific approaches, and other work done on prevention and on site location for a disposal facility.

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Chapter 21

Radioactive Waste Management in Spain Main Activities up to the Year 2000

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21.1 Introduction

In 1995 there were nine nuclear power stations in operation in Spain with a total capacity of 7.4 GW, supplying about 36% of the Spanish electrical energy. At present, spent fuel from the nuclear power plants is stored on site in pools constructed for this purpose. As of the end of 1993, there were 1457 tU of spent fuel.

The estimate of the total volume of high level wastes, spent fuel, that will have to be managed in Spain is up to 11,700 m^3 (40 years of expected life). Total volume of LLW is expected to be about 200,000 m^3 of which about 137,000 m^3 will be from the dismantling of power plants.

The strategy and main activities for the definitive disposal of high level, long lived wastes are given in the General Radioactive Waste Plans (GRWPs). Following a period of intermediate storage of these wastes, their transport and encapsulation, they will be disposed of in a deep geological formation. Transport of the spent fuel will be carried out by ENRESA as the responsible authority, either using the company's own resources or through specialized firms.

Waste conditioning or encapsulation will be carried out at a plant that is planned to be constructed on the same site as the disposal facility. The technique to be used for disposal will be based on a programme of study, research and international co-operation.

Disposal of LLW, will continue in the "El Cabril" facility, which has been in operation since 1992.

The legal framework governing radioactive waste storage facilities and radioactive or nuclear installations in Spain is established by the Nuclear Energy Act 25 of 1964 and the Regulations on Nuclear and Radioactive Installations of 1972.

The "Consejo de Seguridad Nuclear" (CSN), the Spanish Nuclear Safety Council, was constituted in 1980 as an organization existing under Common Law, independent from the Central State Administration. CSN has its own legal standing and corporate assets independent from those of the State, and is the only body in Spain with responsibility in the fields of nuclear safety and radiological protection.

The management of radioactive wastes in Spain is undertaken by "Empresa Nacional de Residuos Radioactivos, S.A." (ENRESA), the Spanish national radioactive waste company, constituted in 1984. Eighty percent of the company is held by the Spanish Centre for Energy, Environmental and Technological Research (CIEMAT), previously known as the "Junta de Energia Nuclear" (Nuclear Energy Council).

21.2 Low and intermediate level wastes

The strategy applied to low and intermediate level wastes continues to be based on a one-to-one relationship between the disposal facility and the wastes themselves. Two major courses of action have been established. The first includes the conditioning, transport and characterization of radioactive wastes and corresponding acceptance criteria, as well as the inspection criteria and procedures required to guarantee compliance. The second includes the design, construction and operation of the disposal facilities.

ENRESA was awarded a Provisional Operating Permit for the Extension to the Nuclear Installation for the Disposal of Solid Radioactive Wastes located in Sierra Albarrana by a Ministerial Order issued on 9th October 1992. As a result of this award, the installation in question, known as El Cabril, will be used over some 20 years for many of the stages involved in managing the LILW generated in Spain, such as conditioning, characterization and disposal. This waste constitutes a new operating stage of special relevance in our country.

21.2.1 Waste Conditioning, Transport, Characterization and Acceptance.

Except in the case of the minor producers, the previous treatment and conditioning of low and intermediate level wastes is the responsibility of the producer, who is obliged to generate packages satisfying the acceptance criteria defined by ENRESA for subsequent conditioning and disposal at the El Cabril facility. In the case of the minor producers, waste treatment and conditioning is carried out at the aforementioned facility.

Transport of the wastes is carried out by ENRESA as the responsible operator, either using its own resources for the removal of wastes generated by the minor producers, or the services provided by specialist companies in the case of conditioned wastes.

The contracts signed between ENRESA and the waste producers include the criteria and technical specifications to be considered in relation to the characterization and acceptance of wastes for subsequent disposal at El Cabril.

A key component in the process of waste quality verification, which to date has been mainly performed abroad, has been the construction in Spain of a Low and Intermediate Waste Quality Verification Laboratory for performance of the corresponding tests (destructive testing, verification, characterization, etc.). This laboratory is part of the El Cabril installations, along with the Disposal Structure Conditioning Plant and other Services.

21.2.2 Disposal of Low and Intermediate Level Wastes

With a view to ensuring the disposal of the low and intermediate wastes produced in Spain, ENRESA operates the El Cabril centre, located in the province of Córdoba; an extension of the works at this facility was completed in 1992.

El Cabril incorporates the most advanced technologies used for this type of installation. Technically, the facility is based on a system of shallow disposal with engineered barriers, similar to the French model. This system guarantees compliance with the necessary safety objectives and criteria, such that there will be no significant radiological impact during the period required for the activity of the wastes to decay to harmless levels. The facility is made up of the following buildings and structures as shown on Figure 21.1:

1. Low and intermediate level waste Conditioning Building, which houses the necessary treatment and conditioning systems (compaction, incineration, manufacturing of hydraulic conglomerant, etc.) for the liquid and solid wastes arising from the application of radioisotopes in medicine, industry, agriculture and research; the solid wastes from CIEMAT, Juzbado Uranium Concentration Plant and the nuclear power plants, and the wastes generated at El Cabril itself as a result of operations.
2. Disposal Structures for the duly conditioned low and intermediate level wastes from the Spanish nuclear and radioactive installations. These structures consist of cells aligned in two rows along two esplanades; it is estimated that their capacity will cover Spain's needs until the end of the first decade of next century (see Fig. 21.2 for layout of disposal platforms).
3. Quality Verification Laboratory where the processes of characterization, testing and control of the characteristics of radioactive packages received or conditioned at the facility are carried out, and for research activities aimed at enhancing the processes of low and intermediate level waste conditioning and characterization.
4. Services and Control Building where industrial safety, reception, technical services, general services, maintenance workshop, concrete container manufacturing and administration are carried out.

The El Cabril facility has been operational since October 1992, when the buildings and structures described above were constructed and the necessary assembly operations and tests were performed.

Up to that date, ENRESA had stored the conditioned low and intermediate level wastes from CIEMAT and the minor producers in the surface modules of the old El Cabril installations. In recent years, these modules have also been used for packages from the José Cabrera, Santa María de Garoña and Ascó nuclear power plants. The other (conditioned) low and intermediate level wastes generated in Spain are temporarily stored at the producers' authorized on-site installations awaiting transfer to El Cabril.

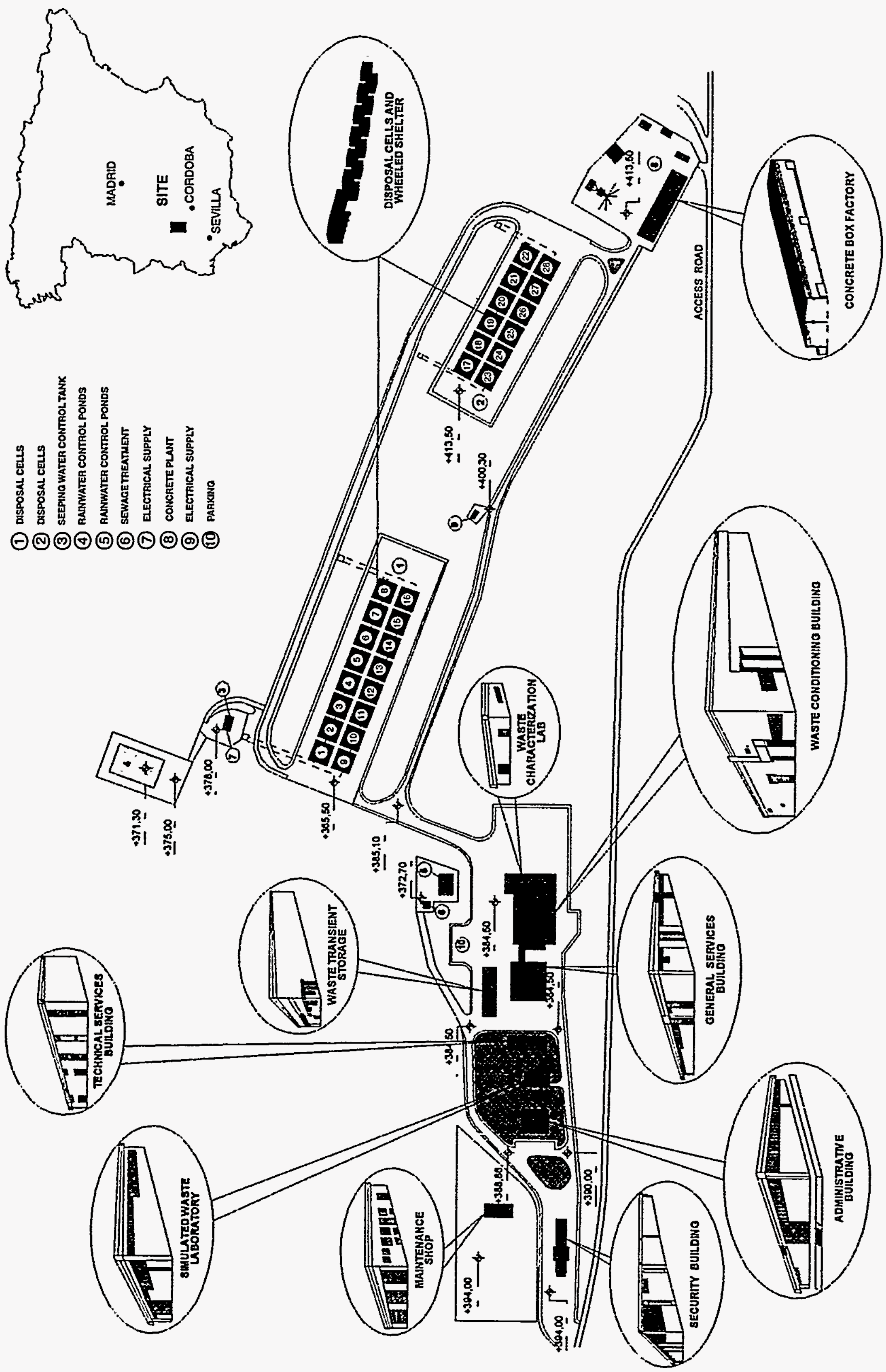

Figure 21.1. General layout of the "El Cabril" LLW disposal facility.

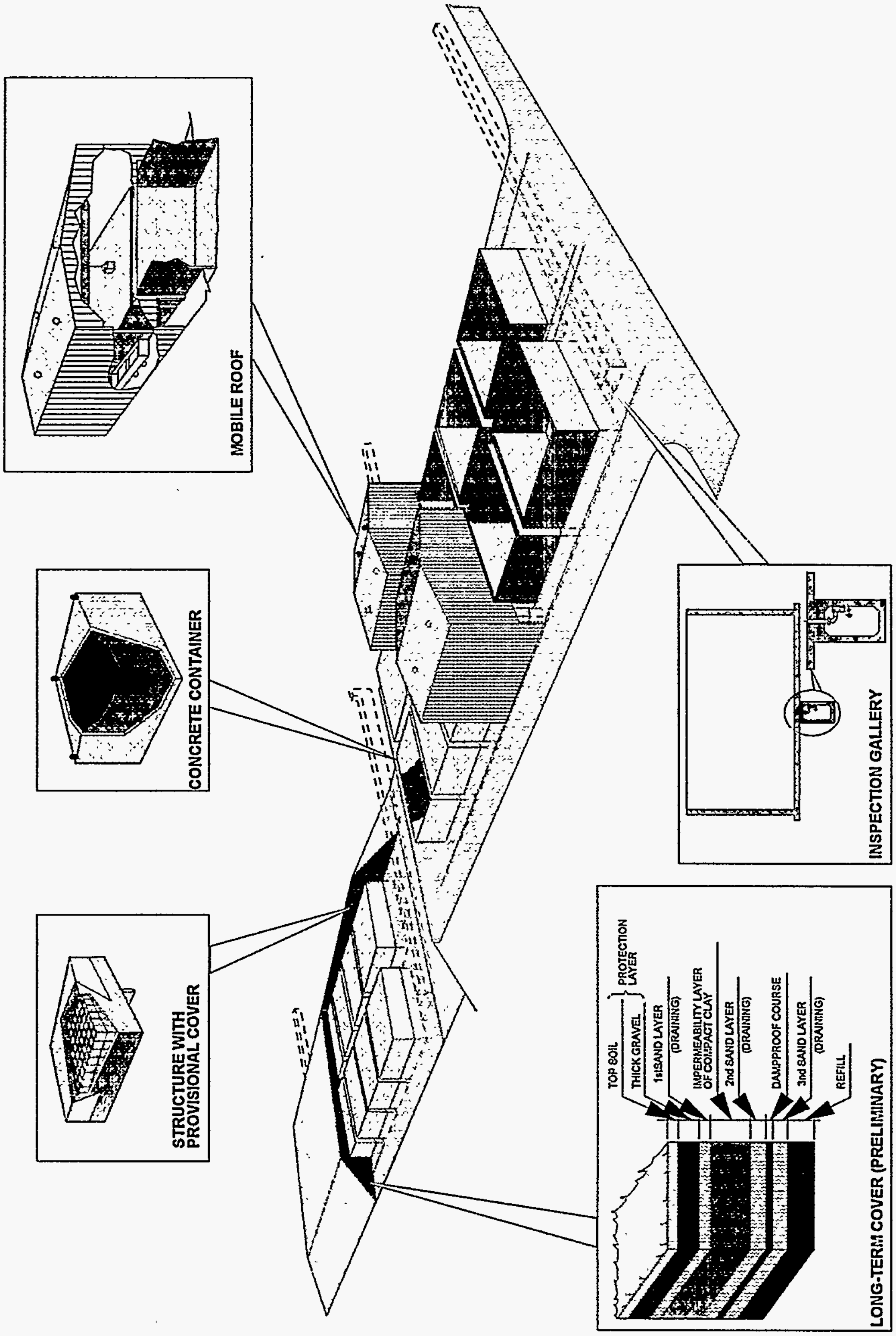

Figure 21.2. Layout of the disposal platforms.

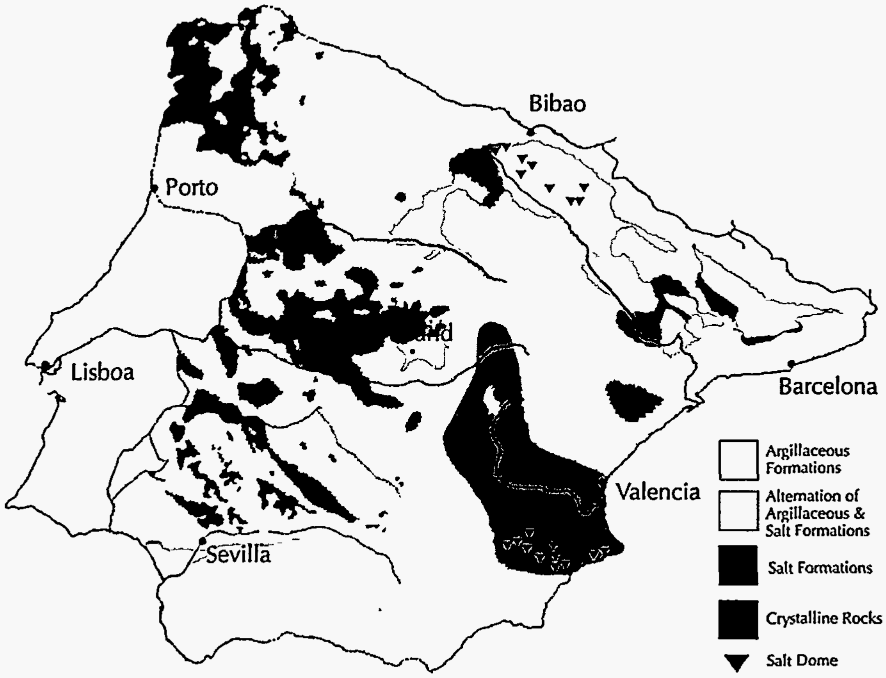

Figure 21.3. European catalogue of geological formations in Spain having favourable characteristics for disposal of HLW.

All this makes Spain one of the few countries, along with France, United States, Sweden, United Kingdom and Japan, to possess an overall capacity for management of low and intermediate level wastes produced by the country's nuclear power plants and some one thousand smaller installations (hospitals, industry, etc.) from a modern environmental point of view.

21.3 HIGH LEVEL WASTES

In Spain, high level wastes are understood to be the non-reprocessed spent fuel from nuclear reactors, the exception being the fuel from Vandellós I NPP, which has been sent to France for reprocessing.

In view of the overall open cycle strategy applied to this fuel in Spain, there are basically two types of high level wastes which will have to be managed: (a) the spent fuel generated by the country's light water reactor nuclear power plants, by far the larger volume, and (b) the vitrified wastes arising from the reprocessing in France of the Vandellós I spent fuel.

Before the definitive disposal of these wastes is accomplished, it is necessary for them to be kept for a period of interim storage in order for prolonged cool down and decay of their isotopic activity to occur.

According to the economic studies carried out, the most reasonable solution for interim storage of the spent fuel, while the nuclear power plants are in operation and taking into account the plant lifetime considered, will be for them to be stored on site at the plants where such spent fuel is generated. This storage will be accomplished either in the plant fuel pool or using dry storage techniques on site. Consequently, the date by which a centralized temporary storage facility for this fuel should be available will depend, fundamentally for the time being, on the time at which the first nuclear power plant dismantling process is undertaken, in other words, on the service lifetime considered for these installations.

The date on which the deep geological disposal facility enters operation will not, however, undergo any variation, regardless of whatever hypothesis is adopted regarding service lifetime.

21.3.1 Search for a Site for Construction of Facilities.

The process of designating a site for disposal of HLW started in 1986 and continues today. The geological media contemplated are granites, salts and clays. A National Inventory of Favourable Formations (IFA Project), has been developed during 1986-7 which confirms the Spanish contribution to the European Catalogue (Fig. 21.3), and a selection process known as the "High Level Regional Studies" (ERA Project) has been drawn up. The site screening in the ERA project has been based on geological, hydrogeological, seismic, environmental and societal data.

The second phase of the process known as the "Study of Favourable High Level Areas" (AFA Project), which covered the period 1990-1995, identifies more than one thousand municipalities with potential capabilities to construct a HLW repository.

In the 4th General Radioactive Waste Plan (December 94) a project of law to arbitrate the procedures to designate the site of surface and underground installations is under consideration by the government. The bill will dictate the means of participation in the ultimate decision of State and other concerned institutions, as well as the general public.

21.3.2 Development of Basic Design for Deep Geological Disposal Facility.

Progress is being made in defining the conceptual design for future surface and underground installations. The aim of this project, which was initiated in July 1990 and is being performed by Spanish engineering firms, in collaboration with Swedish and German organizations, is to perform systems analyses, and to evaluate various detailed disposal concepts and alternatives.

This entire process serves as an important central activity of the R&D programme and is needed to design and construct the disposal facility, regardless of whatever geological medium is chosen.

The first important milestone was the development in 1992, of a preliminary conceptual design for salt and granite formations.

Once ENRESA analyzed the results obtained, a second three-year phase was addressed. Its aim was to get the disposal concept for the three formations ready by the end of 1995, supported by a preliminary safety assessment. Successive later phases are to be undertaken until the selection of the final disposal system project is made.

21.3.3 Acquisition of Technology and Training of teams Required for Characterization of Chosen Site and Construction of Disposal Facility.

Site characterization and disposal system design and verification require a scientific technological support system provided by associated R&D activities. Once human and technical equipment have been selected by previous R&D investigations, a further verification of different methodologies using field studies on different scales will be needed. These studies will provide the necessary information for a long-term performance assessment of the disposal system.

After the final candidate site is selected, the three above mentioned areas of work will be directed towards the same objective which can be summarized as follows:

1. Detailed site characterization by R&D developed and perfected techniques. These will include surface workings, drilling and an underground research laboratory.
2. The previous disposal system design will be adapted by preparing a detailed project to assist in its final construction.
3. The R&D Plan will be aimed at completing the remaining activities, particularly the safety assessment of the chosen site, including specific works to be performed on the site.

All the planned works are to be completed by year 2015 and the final disposal system construction is foreseen to start and finish during the decade of 2020.

21.4 DECOMMISSIONING OF INSTALLATIONS

From the technological and waste production point of view, the most significant aspect of this important management issue in Spain is decommissioning the country's nuclear power plants. In this respect the Vandellós I NPP is of particular importance at this moment in time, with decommissioning of other plants currently in operation constituting a longer term activity.

In spite of the importance of these plants, there are other installations, such as uranium mines, the Andújar uranium mill and the La Haba concentrates manufacturing facility in Badajoz, whose decommissioning will have to be addressed and which are currently in different phases of performance, as described below.

The spent fuel from the Argos and Arbi experimental reactors was transported to the United Kingdom in 1992 for storage and reprocessing; it is foreseen that the waste generated in the process will be returned to Spain. As regards the JEN-1 reactor, dismantling is currently being addressed by CIEMAT; this organization is carrying out a research and development programme in relation to this issue, with participation by Spanish and overseas institutions and financing by the EU. There is also an agreement with the UKAEA for storage and eventual reprocessing of the fuel from this reactor, which was transported to the United Kingdom in 1992.

With regard to the issue of dismantling, special mention should be made of the particularly important question of the declassification of materials as radioactive wastes, since this implies total or partial exemption from the control systems applied to such materials, thus allowing them to be managed by means of methods similar to those used for conventional wastes. Work is currently advancing rapidly at national and international levels with a view to completing detailed development of specific criteria and methodologies for the application of such exemption practices in Spain.

21.4.1 Closed Uranium Mines

As was pointed out in the Third GRWP, ENRESA has carried out a study of the conditions at closed-down mining facilities belonging to the then Nuclear Energy Board, now CIEMAT. As a result of this study, the decision was taken to perform projects at certain of these facilities with a view to restoring the terrain altered by the operations, eliminating rubble tips, refilling quarries, shafts, etc. and in general carrying out whatever corrective measures might be required for the sites to be integrated into their natural surroundings.

The so-called Action Plan for the restoration of closed uranium mines was finalized at the beginning of 1994. Following the corresponding evaluations, work will begin on detailed development of the project at the mines considered to be of interest, as a preliminary step to performance of the field work. According to current forecasts, these tasks will imply specific actions at 2-3 mines; such actions are to be initiated in the last quarter of 1994 and completed during 1995, including the corresponding control procedures.

21.4.2 Andújar Uranium Mill

Authorization for the decommissioning of the Andújar Uranium facility was awarded by Ministerial Order on 1st February 1991, and performance of planned activities began immediately. For performance of the Decommissioning Plan, ENRESA analyzed the technology used in other countries for this type of project and defined the activities to be performed using the USA UMTRAP (Uranium Mill Tailing Remedial Actions Project) programme, which covers 24 installations of this type, as a point of reference.

The proposal includes dismantling of the installations, demolition of buildings and incorporation of the resulting rubble into the mass of tailings, and stabilization of the whole through reduction of banks and construction of a cover providing protection against erosion, diffusion of radon, and infiltration of water.

The design criteria and objectives contemplated relate to the control of dispersion, long-term radiological protection, durability, the cleaning of contaminated soils, the control of radon diffusion, protection for groundwaters and the minimization of long-term maintenance.

The works were completed in May 1994, in accordance with the existing schedule; what now remains to be accomplished is establishment of the corresponding monitoring programme, to be performed over the next 10 years and prior to declaration of the decommissioning of the facility.

21.4.3 Decommissioning of La Haba

The main activities to be accomplished at LaHaba include restoration of the terrain affected by the mining works, by transferring the rubble tips to the mine openings and subsequent replanting, and closure of the Lobo-G plant and the associated tailings dike. This task consists of dismantling the installations and the stabilization and covering of the dike.

21.4.4 Decommissioning of Nuclear Power Plants

Following the final removal from service of the Vandellós I NPP, it was necessary to adjust development of the strategies and technical activities contemplated for this area of management in the first waste plans. These emphasized the fact that this particular problem was a long-term issue and contemplated initiation of total decommissioning (Level 3) of all the Spanish nuclear power plants five years after final shutdown of the reactor.
Based on experience acquired in other countries, especially in France where the technology originated, ENRESA has performed studies aimed at defining the most feasible strategy from a technical and economic point of view, taking into account the specific circumstances of Vandellós I NPP.

The following three possible alternatives were considered:

1. Maintenance of the plant in the final definitive shutdown for an indefinite period of time. (Level 1 dismantling).

2. Dismantling the conventional parts of the plant and active parts other than the reactor and its internals (Level 2 dismantling).

3. Total dismantling, leaving the site in conditions allowing it to be used without any type of restriction (Level 3 dismantling).

To date, no Level 3 dismantling process has been carried out at any commercial plant. Consequently, this alternative may be ruled out for Vandellós I in the short term, owing to the technological, methodological and licensing risks involved. After following a process of study and assessment of various parameters (technological, radiological impact, regulatory, economic, logistics and the volume of wastes to be managed), it was considered that the most feasible strategy for decommissioning the Vandellós I NPP would be immediate dismantling in accordance with alternative 2, followed by a period of waiting, estimated to last 25 years, for completion of total dismantling the remaining parts of the plant in accordance with alternative 3.

The alternative chosen not only represents the most feasible approach from the point of view of both performance and impact on general waste management, but is also backed by French experience in relation to the dismantling of the two units of the Saint Laurent des Eaux plant (SLA 1 and 2). This led to the decision to undertake a Level 2 dismantling followed by total dismantling (Level 3) following a suitable waiting period, estimated at between 25 and 30 years.

At Vandellós I NPP, the programme of activities to be performed prior to dismantling is being coordinated with the fuel removal activities carried out by HIFRENSA, such that during 1995, the detailed engineering project and licensing process will have been completed. In this respect, ENRESA submitted a dismantling and decommissioning project to the Ministry of Industry and Energy in May 1994, for its approval. This project contemplates partial dismantling of the facility (Level 2), which will make it possible to determine the most suitable period of waiting prior to initiation of the total dismantling process. It is estimated that four years will be required for completion of the partial dismantling process contemplated in this project, as from the date of initiation.

As regards the other light water reactor nuclear plants currently in operation, consideration is currently still being given, from the point of view of calculation and planning, to the alternative of undertaking complete dismantling (Level 3). This process is to be initiated between four and eight years after final shutdown of the plant.

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CHAPTER 22

PROGRESS TOWARDS A SWEDISH REPOSITORY FOR SPENT FUEL

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Abstract. Nuclear power is producing electricity for the benefit of society but is also leaving radioactive residues behind. It is our responsibility to handle these residues in a safe and proper manner. The development of a system for handling spent fuel from nuclear power plants has proceeded in steps. The same is true for the actual construction of facilities and will continue to be the case for the final repository, for spent fuel, and other types of long-lived wastes. The primary objective in constructing the repository will be to isolate and contain the radioactive waste. In case the isolation should fail for some reason, the multibarrier system shall retain and retard the radionuclides that might get in contact with ground water. A repository is now planned to be built in two steps, where the first step would include deposition of about 400 canisters with spent fuel. This first step should be finished about 20 years from now and be followed by an extensive evaluation of the results from not only this particular step but also from the development of alternative routes before deciding on how to proceed. A special facility to encapsulate the spent fuel is also required. Such an encapsulation plant is proposed to be constructed as an extension of the existing interim storage CLAB. Finding a site for the repository is a critical issue in the implementation of any repository. The siting process was started a few years ago and made some progress but is by no means yet completed. It will go on at least into the early part of the next decade. When the present nuclear power plants are about to be shutdown, there should also be facilities in place to take permanent care of the long-lived radioactive residues. Progress in siting will be a prerequisite to success in our responsibility to make progress toward a safe permanent solution of the waste issue.

22.1 INTRODUCTION

Sweden has twelve nuclear power reactors with 10,000 MWe electric capacity. These reactors are producing some 70 TWh per year and from that production arises about 250 tonnes of spent nuclear fuel. Up to 2010, it is estimated that the cumulative amount of fuel will be some 8,000 tonnes (uranium weight). The responsibility for taking care of this fuel rests with the owners of the nuclear power plants in accordance with the principle that "the producer is responsible". The owners have given the mission to the Swedish Nuclear Fuel and Waste Management Co. (SKB) to execute this responsibility for them.

The progress of the Swedish waste management programme is closely monitored by the government and by the pertinent authorities in Sweden. According to the law, SKB has to submit a programme for research, development and all other necessary measures in order to be able to handle and finally dispose of the spent fuel and other radioactive wastes arising from the operation of the nuclear power plants. Such a programme must be submitted at three year intervals and is then scrutinized by the authorities and a broad set of reviewers before the government decides on the programme. In the past, SKB has submitted several such programmes (SKB 1986, SKB 1989, SKB 1992, SKB 1995a). The latest programme is still under review by the government.

As the implementation of the programme proceeds and reaches the siting and construction phases, the same authorities will be responsible for evaluation of the safety and giving stepwise permission to proceed towards completion of the system.

22.2 STEPWISE DEVELOPMENT

The development of a system for handling the spent fuel has proceeded in steps. The first steps were taken during the 1970s when a parliamentary committee proposed the construction of a central interim storage facility for spent nuclear fuel and to initiate research for disposal of high level radioactive wastes deep in the crystalline bedrock in Sweden (Aka 1976). The research took a great step forward with the KBS-project which was established in late 1976 in response to a new law. This was the stipulation law, which required that the nuclear power plant owners work out a plan for handling and final disposal of high level waste from spent fuel before the last six reactors of the power programme were allowed to start operation. These plans became known under their abbreviated names KBS-1, KBS-2 and KBS-3 (SKBF 1977, SKBF 1978, SKBF 1983, respectively).

KBS-1 addressed the disposal of vitrified waste from reprocessing, whereas KBS-2 and KBS-3 described the disposal of unreprocessed spent nuclear fuel. All three studies included a period of interim storage before the final disposal; a period of 30 - 40 years was considered appropriate in order to decrease the thermal load on the repository. After about 40 years, the residual decay heat in the fuel will have decreased by about 90 % in comparison with one year old spent fuel. A further equally important consideration in favour of interim storage is the creation of flexibility and buffer capacity in the management system for the spent fuel.

In the 1980s, the Swedish programme was focused on final disposal of the spent fuel without reprocessing. There were several reasons. The nuclear power programme was limited to twelve reactors. The prices for natural uranium and enrichment services stabilized or even decreased, whereas the prices for reprocessing services and Pu-recycle continued to increase. Thus, direct disposal got an economic advantage. The concern for nuclear proliferation, as a consequence of increased handling of plutonium, created a political resistance against reprocessing and Pu-recycle. The development of the breeder slowed down.

Several alternative methods for final disposal of spent fuel in the Swedish crystalline bedrock were studied and evaluated as a part of the broad R&D-programme, which continued based on the KBS-reports. In 1992, SKB concluded that a method similar to the one described in the KBS-3-report would be best suited for use, at least for the first step, in implementation of a deep repository. In general this conclusion was accepted by the authorities although some reviewers considered the choice to be premature.

22.3 STEPWISE CONSTRUCTION

An interim storage facility, CLAB, was constructed in the early 1980s at the Oskarshamn nuclear power plant. It was put in operation in 1985. CLAB has at present a capacity of 5,000 tonnes and can easily be expanded to meet future demands. About 2,300 tonnes of spent fuel was in storage at the beginning of 1996.

Following the evaluation of alternative methods in 1992, SKB decided to start the implementation process for the first steps in building a deep repository for spent nuclear fuel. This comprises the siting and basic design of an encapsulation plant for the fuel and of a deep repository. The first stage of the repository is planned for about 400 canisters or 800 tonnes of fuel and should start operation in 2008 (Fig. 22.1). The encapsulation plant is planned for filling one canister per day.

After the first stage has been completed, a thorough evaluation will be made both of the experience gained from the first stage and from development of other, alternative treatment and disposal methods that have been studied and/or applied in Sweden or elsewhere. The opportunity to change the route or even to retrieve the canisters that have been deposited will be available. This strategy thus provides an approach where irrevocable decisions must not be made until all aspects of the repository implementation have been fully demonstrated.

The implementation of the first stage will also proceed in steps with siting, basic design and supplementary R&D during the 1990s, with construction during the main part of the next decade and the first stage operation and evaluation during the 2010s. The stepwise approach is thus a key element in the planning and implementation for a repository.

22.4 SAFETY APPROACH FOR A DEEP REPOSITORY

The safety of a deep repository is dependent on the radiotoxicity and on the accessibility of the waste. Both these properties are time functions. Thus, the safety of the repository has to be assessed as a function of time. There will always be a fundamental uncertainty in the prediction of the future behavior of any system and the uncertainty may increase with time. The Swedish Radiation Protection Institute has discussed the influence of the time horizon on the safety assessment and radiation protection (SSI 1995). They conclude that:

- Particularly great attention should be given to describing protection for the period up to closure of the repository and during the first thousand years thereafter, with a special focus on nearby residents.
- The individual dose up to the next ice age, i.e. up to about 10,000 years, should be reported as a best esti-

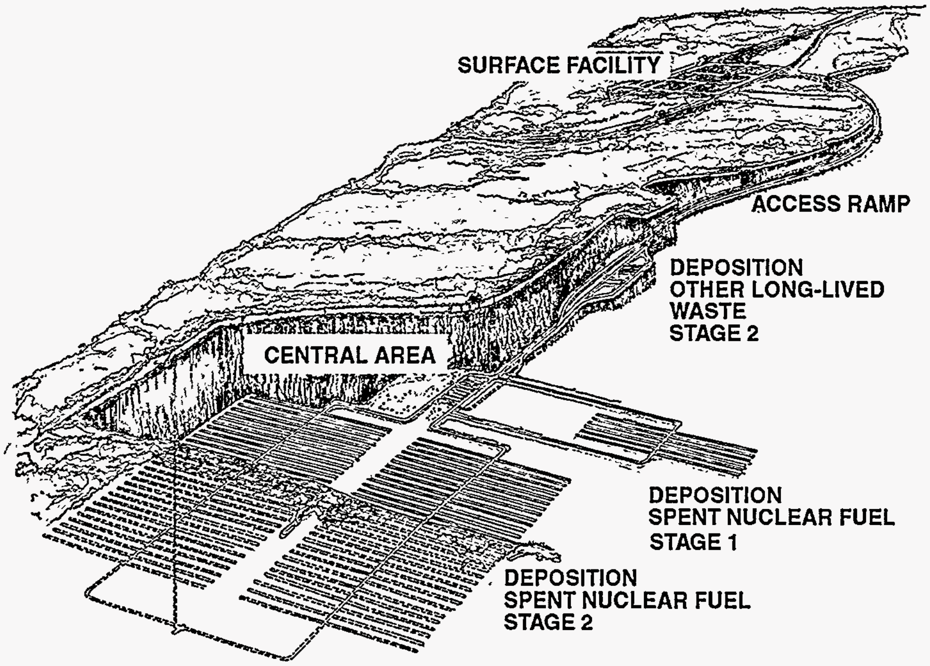

Figure 22.1. Deep repository showing schematic layout of stage 1 and stage 2.

mate with an estimated margin of error. Environmental protection should be described for the same period of time.

- For the period from the next ice age onward, qualitative assessments should be made of what might happen with the repository, including scenarios taking into account the risk of increased releases.

The safety of a repository is achieved by the application of three principles:

- Level 1 - Isolation. Isolation enables the radionuclides to decay without coming into contact with man and his environment.
- Level 2 - Retardation and retention. If the isolation is broken, the quantity of radionuclides that can be leached and reach the biosphere is limited by:
  - very slow dissolution of the spent fuel;
  - sorption and very slow transport of radionuclides in the near field; and
  - sorption and slow transport of radionuclides in the bedrock.
- Level 3 - Recipient conditions. The transport pathways along which any released radionuclides can reach man are controlled to a great extent by the conditions where the deep groundwater first reaches the biosphere (dilution, water use, land use and other exploitation of natural resources). A favourable recipient means that the radiation dose to man and the environment is limited. The recipient and the transport pathways are, however, influenced by natural changes in the biosphere.

The safety functions at levels 1 and 2 are the most important and the next-most important. They are achieved by means of requirements on the properties and performance of both engineered and natural barriers and on the design of the deep repository. Within the frames otherwise defined, a good safety function at level 3 is also striven for by a suitable placement and configuration of the deep repository.

22.5 DEEP REPOSITORY

The isolation of the spent nuclear fuel from the biosphere is achieved by encapsulating the fuel in a canister with good mechanical strength and very longlived resistance against corrosion. The conceptual design adopted is a copper canister with a steel insert. The copper provides a very good corrosion resistance in the geochemical environment foreseen in a deep repository in Sweden. The steel insert provides the mechanical protection needed. Each canister contains about 2 tonnes of spent fuel. The canisters are placed in deposition holes drilled from the floors of tunnels at about 500 m depth in the crystalline, granitic bedrock (Fig. 22.2). Each can-

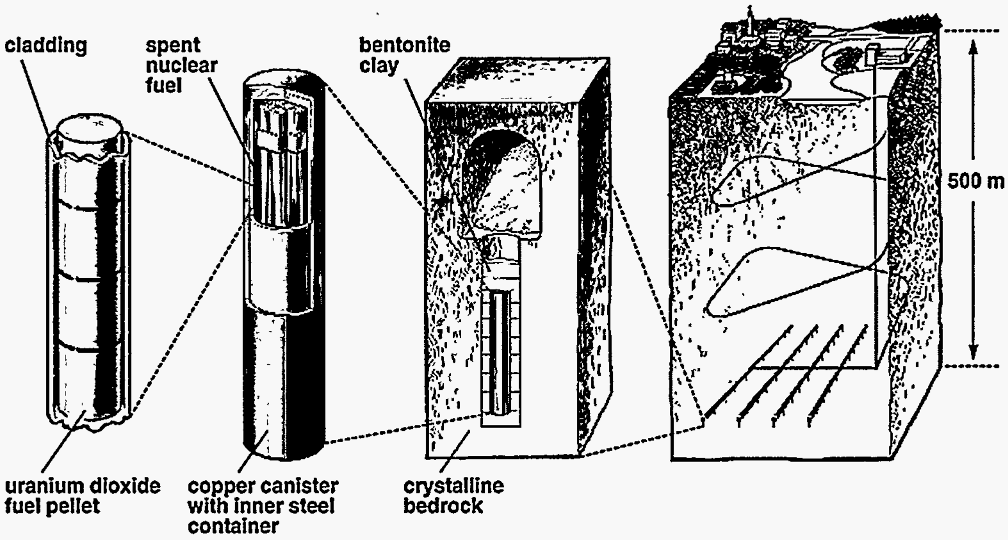

Figure 22.2. Deep repository in accordance with the KBS-3 concept.

ister is surrounded by blocks of compressed bentonite. When the bentonite absorbs water from the surrounding bedrock it will exert an intense swelling pressure and completely fill all void space in the near vicinity of the canister with bentonite clay. The clay barrier will contribute to the isolation by preventing or delaying dissolved corrosive species that may exist in minor amounts in the ground water to reach the canister. The clay will also provide some mechanical protection for the canister. The tunnels will eventually be backfilled by some material like a mixture of crushed rock and bentonite.

A repository for all spent fuel from the present Swedish programme should have a capacity of about 8,000 tonnes or 4,000 canisters. In addition it should be able to deposit other types of long lived wastes at the same site. This means that the underground facilities will need some 30 - 40 km of tunnels and cover an area of about one km^2. The surface facilities at the repository site will require an area of about 0.2 km^2.

SKB has started the planning work for a deep repository by preparing plant descriptions. These provide examples of possible ways to design the repository with its buildings, land areas, rock caverns, tunnels and shafts. They also contain requirements on, and principles for, the various functions of the repository. To a large extent, the construction and operation of the facility can be based upon experience and proven technology from nuclear installations and underground rock facilities. Special attention is given to: the impact of the excavation work on surrounding rock, methods for preparation and installation of the buffer bentonite blocks, and the technology for backfilling and sealing.

22.6 ENCAPSULATION OF SPENT NUCLEAR FUEL

Another necessary facility where the planning work has started is a plant for encapsulating the spent nuclear fuel. The intention is to expand the existing interim storage facility, CLAB, at Oskarshamn with such a plant. The plant will take fuel assemblies from the underground storage pools, dry them, transfer them to canisters made of copper with a steel insert, change the atmosphere to inert gas, put lids on the canisters and seal the lids by electron beam welding. The quality of the filled and sealed canisters will be inspected by non-destructive examination (NDE) methods - ultrasonic and radiographic - before shipping to the repository.

Each canister will contain 12 BWR fuel assemblies or 4 PWR assemblies. The copper thickness will be about 50 mm and the steel thickness also about 50 mm (Fig. 22.3). The copper thickness shall be enough to prevent corrosion from penetrating the canister during the time when the spent fuel radiotoxicity substantially exceeds what one would find in a rich uranium ore. The combined thickness of steel and copper should be enough to prevent any significant radiolysis of water outside the canister after deposition in wet bentonite clay. The steel insert is designed to withstand the normal mechanical loads that will prevail on the canister in the repository such as hydrostatic pressure and the bentonite swelling

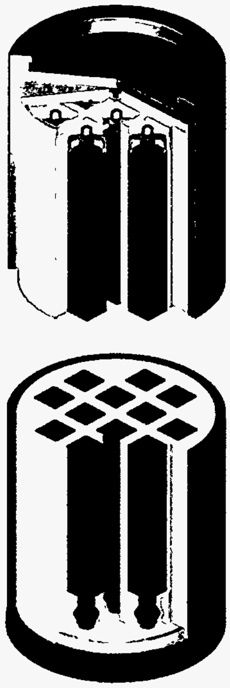

Figure 22.3. Copper canister with steel insert.

pressure. The total weight of a canister with fuel will be about 25 tonnes. In total some 4,000 canisters will be required for the spent fuel arising from the Swedish reactors up to 2010.

The design of the steel insert is still under evaluation. The present reference concept is a cast insert with thick steel walls between each fuel assembly. This gives a good mechanical stability as well as providing adequate protection against criticality in the unlikely case that the canister, at some unspecified future time, should be filled with water.

The fabrication of copper canisters of the size needed is by no means an industrially available technology. The seal welding technology has recently been demonstrated on a laboratory scale in work sponsored by SKB at The Welding Institute in UK. Full scale canisters have also been fabricated on the laboratory scale. In order to make key technology more mature, SKB has decided to create a laboratory for encapsulation technology at the former shipyard in Oskarshamn. This laboratory will be ready sometime during 1997 and will primarily be devoted to further development of the seal-welding process and the NDE-methods.

The design of the encapsulation plant is in progress. The main contractors for the design are BNFL for the key hot cell parts and ABB Atom for the other systems. The work is at present directed towards final specifications for the above mentioned laboratory as well as the development of an Environmental Impact Statement and a Preliminary Safety Report forming a basis for an application for a siting and construction permit. The application is planned for submission in early 1998.

22.7 REPOSITORY SITING

The most crucial part of the development of a deep repository is the siting process. SKB started geological site investigations at an early stage in the programme. Throughout the 1970s and 1980s, so called study sites were investigated at some 10 locations scattered from the southern to the northern part of the country. The investigations included measurements in boreholes as deep as 1000 m, as well as geophysical measurements from the surface. The main conclusion from these site investigations was that there are many places in the Swedish bedrock that provide conditions which are suitable for siting a repository at a depth of about 500 m. This implies that the safety requirements for a site can be met at many places and that factors other than safety can also be decisive in siting.

One such factor of importance is the acceptance by authorities and local residents. After the presentation of the RD&D-programme 92, SKB got in contact with several municipalities in various parts of Sweden. These contacts led to the proposal to start so called feasibility studies for the municipality in order to more clearly define the possibilities and consequences of siting a repository in the municipality. The intent was that the municipality as well as SKB should get a comprehensive set of documentation on which to base any decision of further, more detailed studies. A prerequesite was that the feasibility study should be based on existing geological data; no drillings and new measurements would be included. The discussions resulted in feasibility studies in Storuman and Malå in northern Sweden, Lappland (Fig.22.4).

The study for Storuman was published in February of 1995, (SKB 1995b) and that for Malå in March 1996 (SKB 1996). In both municipalities, two fairly large areas - 50-100 km^2 - were identified to be of interest for further investigations as potential host formations for a repository. However, in September of 1995, Storuman had a referendum on whether to permit further investigations for a repository site in the municipality, and the outcome was more than 70 % no-votes. This means that

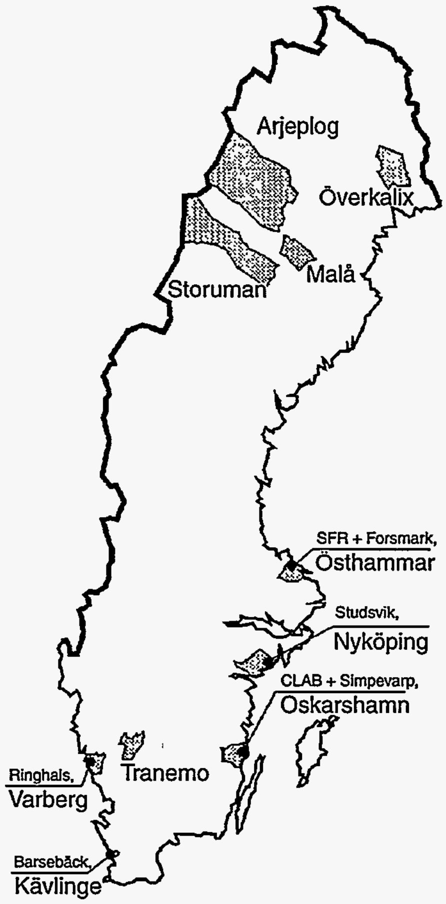

Figure 22.4. Location of some municipalities in Sweden.

SKB's work at Storuman has ceased. The study in Malå is now being reviewed by the municipality with the help of an independent group of experts.

SKB also performed a general study of five municipalities: Östhammar, Nyköping, Oskarshamn, Kävlinge and Varberg, where nuclear facilities are already located. The conclusion from this study was that there was a clear interest to continue with specific feasibility studies for the three first mentioned municipalities. There was also some interest for a study in Varberg, although the existing geological data for that area are rather meager and must be supplemented. As a result, feasibility studies are now going on in Östhammar and Nyköping whereas the proposal is still being evaluated by Oskarshamn. Varberg has declined a feasibility study.

SKB has also had fairly extensive discussions with three other municipalities: Överkalix, Arjeplog and Tranemo. These have finally resulted in negative answers from the municipalities mainly due to strong local opposition. In a few other cases, the negative answers were given at an early stage.

In 1995 SKB published a General Study '95 (SKB 1995c) which provides background material and gives an overview of some important siting factors on a scale covering the whole of Sweden. One main conclusion from this study is a confirmation of previous observations that it is not feasible to identify interesting areas on such a coarse scale. It is, however, possible to identify some major areas like Gotland, Skåne and the mountain range at the border with Norway where the geological and other conditions are such that it is of less interest to look for a site.

Based on the General Study '95, SKB will continue to study some parts of Sweden on a regional scale in order to identify areas of possible interest. The ambition of SKB is to make feasibility studies of some 5 - 10 municipalities as a basis for selecting at least two sites for investigation including extensive drilling as well as geophysical, geohydrological and geochemical measurements. These investigations will give the necessary data base to seek permits to enable detailed site characterization including construction of tunnels and/or shafts down to repository depths. The government has concluded that such a detailed investigation means that part of the construction of the future repository facility will actually start. This means that a siting permit according to the Act on Management of Natural Resources must also be accompanied by a permit according to the Act on Nuclear Activities.

Critics of SKB claim that the siting process followed by SKB is non-scientific and unsystematic; some even claim that it violates the Swedish environmental protection act. The siting process was, however, outlined in detail in the supplement to RD&D-programme 92, which after a rather extensive review was approved by the authorities and by the government. The fourth paragraph of the environmental protection act says: *For an environmentally hazardous activitity, one shall select such a site that the purpose can be achieved at a minimum of impact and inconvenience and without unreasonable costs.* The process followed by SKB for finding a site for the deep repository is fully in line with this paragraph; the strategy is to find a site where the purpose to construct a safe deep repository can be achieved and then at first hand look at sites where there is some local interest to consider receiving the facility. Thus, one should minimize the inconvenience to areas where there

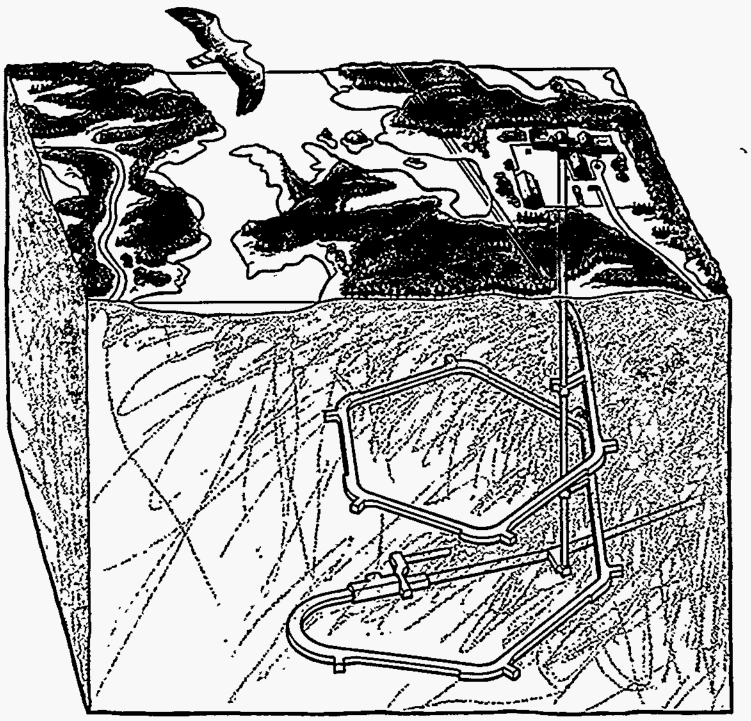

Figure 22.5. General layout of the Äspö hard rock laboratory.

is no such interest. At the same time, the authorities want to have a reasonably broad base for judging that there are no other sites which would obviously be better choices than the one finally selected for an application to construct the repository.

22.8 ÄSPÖ HARD ROCK LABORATORY

In order to prepare for the siting and construction of a deep repository, SKB has built the Äspö Hard Rock Laboratory. The planning of this facility started some 10 years ago in 1986. The work at the laboratory has proceeded in three stages: planning and site investigations, construction, and operations. The first two stages have now been completed and the operational stage has started. A basic objective in the planning of the laboratory was to create a facility for research and development in a realistic and unperturbed environment at a depth planned for the future repository.

The Äspö HRL is designed to meet the requirements on R&D. The underground construction starts with a tunnel from the site of the Oskarshamn nuclear power plant heading north down to about 220 m depth under the island of Äspö. The tunnel then goes down in a spiral with a radius of some 150 m down to 450 m depth. The total length of the tunnel is about 3,600 m. The last 400 m were excavated by a Tunnel Boring Machine (TBM) as opposed to the first part that was drilled and blasted. The cross section of the tunnel is about 25 m^2 (Fig. 22.5).

Overall objectives for the research conducted at Äspö are to:

- increase the scientific understanding of the safety features and function of a repository;
- develop and test technology that will simplify the disposal concept and decrease costs while retaining quality and safety; and
- demonstrate the technology that will be used for disposal of spent fuel and other long-lived wastes.

When the Äspö-project started in the late 1980s, the work program was formulated in several stages. The goal of the first stage was to verify that investigations on the surface and through boreholes from the surface will provide enough data on the important properties related to safety of the bedrock at the repository level. The comprehensive site investigation programme, conducted before the start of construction, was the basis for predictions on geological, geohydrological and geochemical data and behaviour of the rock at depth. These predictions were then compared with observations and measurements performed during the tunnel construction stage. A general conclusion concerning the first stage goal is that the methods available for site investigations are well suited to obtain data and information on the bedrock at a given site. These data can be used to select the proper volume of rock needed for a repository and to make the safety assessment needed for obtaining a permit to construct tunnels and/or shafts for starting detailed site characterization.

The goal of the second stage was to develop the methodology for such detailed site characterization. During the construction of the laboratory, considerable experience has been gained in how the detailed studies of the host rock can be conducted, and a good basis has been laid for the actual work to be made at a repository site in the future.

The other stages were related to improved understanding of the natural barrier - the bedrock - and to a demonstration of the technology to be used in the repository. Some of the tasks addressing these goals have already been performed, whereas others are part of the ongoing experimental and investigation programme at Äspö HRL. A series of Tracer Retention Understanding Experiments (TRUE) is aimed at increasing the knowledge on the capacity of the bedrock to retain and retard the transport of radionuclides in fractured rock. Experimental studies are being carried out to determine how and at what rate the oxygen, present in the repository at closure, is consumed by reactions with the rock. A special borehole laboratory - CHEMLAB - has been developed. It permits chemical experiments to be conducted under repository-like conditions with respect to groundwater composition and pressure. Such experiments will give data to verify models *in-situ* and check the data used to assess the dissolution of radionuclides, the fuel corrosion, the sorption on rock surfaces, the diffusion in buffer materials etc. A full-scale (in-active) prototype repository is planned to be built at Äspö. It will provide the opportunity to simulate all stages in the deposition sequence in a realistic environment. It will also be possible to observe the simulated repository several years in advance of depositing the first (active) canister in the final deep repository.

The work at Äspö has attracted a large international interest, and at present eight foreign organizations from seven countries are participating in the Äspö programme. Several of the experiments are conducted with very active participation by scientists from the foreign participants. This provides a good mechanism for further strengthening the scientific quality of the work and gives all participants access to a broad international forum for discussing the planning, execution, evaluation and interpretation of the experiments.

22.9 CONCLUDING REMARKS

The implementation of a deep repository for spent nuclear fuel is a very lengthy and tedious process in today's society. Considering the time scale intended for the isolation of spent fuel, it is of course still a short time period. In comparison with many other more common projects, however, it is unusual and demanding for all those involved. It extends over several decades and must proceed in steps, where each step requires careful planning. This stepwise progress is a key element. It must be stressed that no step should really be irrevocable; it should always be possible to step back and reconsider and even take another route.

The siting of a deep repository in Sweden is now in progress. To complete this process, efforts will be needed not only from the responsible implementing organization but from all parties involved such as the safety authorities, affected local authorities and political bodies. Building of confidence and trust is a key element in the process. Nuclear power is producing electricity for the benefit of society, but it is also leaving radioactive residues behind. It is our responsibility to handle these residues in a safe and proper manner. When the present nuclear power plants are about to be shut down, there should also be facilities in place to take permanent care of the long-lived radioactive residues. This is the responsibility of our generation which has benefited from the electricity produced. It will be up to the following generations to decide on how to use, extend or change the system we have provided. In this way we can take care of our responsibility without depriving future generations of their possibilities to take their own actions. Considering the existence of long-lived radioactive wastes, a deep repository for such residues will, however, be an asset for society.

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CHAPTER 23

HIGH LEVEL RADIOACTIVE WASTE MANAGEMENT IN SWITZERLAND: BACKGROUND AND STATUS 1995

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23.1 BACKGROUND

Switzerland is a small country, with limited natural resources (other than hydro power) and must import about 80 % of its primary energy needs (predominantly petroleum products). Electricity covers about 20 % of energy demand; about 40 % of the electrical energy is supplied from nuclear plants, with almost all the rest being generated by hydropower.

Nuclear power production is the main source of Swiss radioactive wastes, although wastes arise also in medicine, industry and research. Switzerland currently has 5 nuclear power plants (pressurized water reactors and boiling water reactors) with a total capacity of 3 GW(e). The spent fuel, containing most of the waste radionuclides produced by fission, may be prepared for direct disposal or reprocessed to recover useable uranium and plutonium, with the resulting wastes being immobilized in glass blocks. To date, Swiss disposal planning has focused on waste returned from foreign reprocessing plants but, currently, the preferred strategy of the utilities is to keep open both options (reprocessing or direct disposal). The prime reason for originally choosing the reprocessing route was to optimize use of resources; the current arguments against reprocessing are primarily economic.

Nuclear power and its future role in the nation's energy mix is a controversial issue. The initial widespread acceptance has been replaced to a significant extent by uncertainty or even opposition; this led in 1990 to the adoption by popular referendum of legislation placing a 10 year moratorium on expansion of nuclear power. The waste disposal issue, as will be detailed in the following section, plays a prominent part in the debate on nuclear energy. There is a strong incentive for those responsible for waste management to ensure that continuing progress is made towards development and implementation of an integrated waste management strategy.

23.2 LEGAL, REGULATORY AND ADMINISTRATIVE ISSUES

The Atomic Law of 1959 clearly placed the responsibility for nuclear waste disposal with the producer of the waste. The Ruling of 1978 further stipulates that *"The general license for nuclear reactors will be granted only when the permanent, safe management and final disposal of radioactive waste is guaranteed."* This Ruling was extended by the Government to existing reactor operating licenses and this led to preparation of a special Project Gewähr (PG85) which, as described below, was submitted to the Government for review in 1985[1].

The safety conditions which the final repositories must satisfy are defined in the Guideline R-21 (1980, revised 1993) of the Nuclear Regulatory Authorities. Three protection objectives are defined:

- The repositories must ensure the safety of human beings and the environment from any harmful effects of ionizing radiation. Accordingly, the central point is Objective 1, which states: "Radionuclides which can be released into the biosphere from a sealed repository as a consequence of realistically assumable processes and events may not at any time lead to individual doses which exceed 10 mrem (0.1 mSv) per year." This objective is ambitious not only because of the absence of any time limit for demonstration of compliance but also because of the comparatively low levels of radiation doses permitted. For comparison, the natural radiation exposure of the Swiss population results in an average radiation dose of about 140 mrem per year, with a range of approximately 100 to 300 mrem per year.

- Objective 2 provides a quantitative risk level for judging the consequences of low-probability scenarios: "The individual radiological risk of fatality from a sealed repository subsequent upon unlikely processes and events not taken into consideration in Protection Objective 1 shall, at no time, exceed one in a million per year". The direct radiological risk of fatality from a scenario is thus multiplied by the estimated probability of the scenario occurring and this product should not exceed one in a million per year when summed over all such scenarios. For comparison, the dose limit of 0.1 mSv per year corresponds to a nominal risk of fatality of 5 in a million per year.

- Besides the safety aspects, the Guideline R-21 reflects the understanding that the responsibility for disposing of radioactive waste lies with today's beneficiaries of nuclear power and should not be passed on to future generations. This is expressed in Objective 3: "A repository must be designed in such a way that it can at any time be sealed within a few years. After a repository has been sealed, it must be possible to dispense with safety and surveillance measures." Once the repository has been sealed, it must thus be possible to "forget" the radioactive waste in the sense that it should not be necessary for future generations to concern themselves with it. There is thus no requirement for monitoring or retrievability of the waste.

In addition to the requirements formulated in these three protection objectives, non-nuclear regulations must also be taken into consideration; these include international law, district planning, environmental protection; and nature conservation.

As noted above, the producers of nuclear waste are responsible for waste management (for all waste categories). Hence the electricity supply utilities involved in nuclear power generation and the Swiss Confederation (which is directly responsible for the waste from medicine, industry and research) joined together in 1972 to form the "National Cooperative for the Disposal of Radioactive Waste" (Nagra). Nagra is responsible for the disposal and, if required, pre-emplacement conditioning of wastes. The responsibility for spent fuel reprocessing and transport, for the waste conditioning at power plants and for interim storage remains directly with the utilities. In 1994, a separate organization was founded to actually construct and operate a L/ILW repository at a site selected by Nagra, the Genossenschaft für die Nukleare Entsorgung Wellenberg (GNW).

23.3 CHARACTERISTICS AND EVOLUTION OF THE SWISS NUCLEAR WASTE DISPOSAL PROGRAMME

Since the founding of Nagra in 1972, work has been carried out on the development of disposal concepts and identification of potential sites for such facilities. Working on the principle of the multi-barrier concept, the requirements for packaging, engineered structures and geological isolation were derived for different types of waste. Two separate geological repositories are planned[2]; one for low-level radioactive wastes and shorter-lived intermediate-level wastes (L/ILW) and another for high-level wastes (HLW) and intermediate-level wastes containing higher concentrations of long-lived alpha- emitters ("TRU").

Highest priority at present is allocated to the L/ILW repository which is intended to be implemented in horizontally accessed rock caverns with some hundreds of metres of overburden. An extensive site-selection procedure resulted in 1993 in the nomination of Wellenberg in Central Switzerland as the preferred repository location. More detailed site-characterization work, to form the basis of the application for construction and operation permits, is now ongoing. The principal for the development of Wellenberg as a repository site has been accepted in public Referenda at the community level. It is also supported by the federal authorities. At the cantonal level, however, there has been opoposition, leading in 1995 to a popular vote which produced a narrow majority against the currently proposed project. Nevertheless, current planning assumes that, following appropriate amendments to the project, the L/ILW repository should be operational early next century.

For the present limited nuclear programme in Switzerland, operation of all plants for a 40 year lifetime will result in around 3000 t of spent fuel. If all of this were reprocessed abroad, the resulting volume of vitrified waste returned would only be around 500 m^3, although several thousand cubic metres of additional L/ILW could also be returned (depending on the contract with the reprocessor). It is planned to store HLW for at least 40 years in order to reduce the thermal loading of the repository, so that ample time is available for project development. A centralized facility for dry cask storage of spent fuel and of vitrified HLW and for other reprocessing wastes will be constructed before the turn of the century by the ZWILAG organization, a daughter company of the utilities.

Implementation of a HLW repository will not take place in Switzerland before the year 2020, and there are sound economic arguments for delaying this date even further. Nevertheless, there is strong pressure from the public and the government - and a strong will on the part of the waste producers - to move the project ahead as quickly as possible, at least up to the level of demonstrating the feasibility of construction of a safe repository at a potential site.

Site selection is very much constrained by the small size of Switzerland and by its relatively active tectonic setting. The current geological consensus is that the orogeny which built the Swiss Alps is still continuing and there is still net uplift in this area of ~1-2 mm/year (which is equivalent to 1-2 km in the million year timescale which is considered for HLW safety analysis!). Excluding alpine areas and other complex geological structures associated with the Jura mountains and the Rhine Graben leaves only a limited area in Northern Switzerland which would be potentially suitable. Within this area, two host rock options are considered - either the crystalline basement or one of the overlying, low permeability sediment layers.

The current conceptual repository design was developed taking into account the potential host rocks, the very low volumes of HLW expected and the government requirement for an early, convincing demonstration of safety of waste disposal as a condition of extending reactor operating licenses. These factors together led to designs which are certainly robust (or even overdesigned) and are not optimized in an economic sense. Accordingly, although estimates of absolute costs for the small size repository required are comparable to those from other countries, the costs per unit waste volume (or per kWh of nuclear electricity) are relatively high. Optimization of designs would, therefore, clearly be an important objective before moving to an implementation phase. The concept, illustrated in Figure 23.1 for the crystalline host rock option[3], has the following features:

1. extremely deep disposal (about 1 km below surface) in a carefully-constructed facility;
2. in-tunnel emplacement of HLW waste packages in a geologic medium whose principal roles are to limit water flows and to ensure favorable groundwater chemistry;
3. very massive engineered barriers; in addition to the vitrified waste in its steel fabrication canister, a 25 cm thick steel overpack is envisaged which is surrounded by more than one metre of highly compacted bentonite clay (Fig. 23.2); and
4. co-disposal of TRU in silos or in caverns in a separate part of the repository.

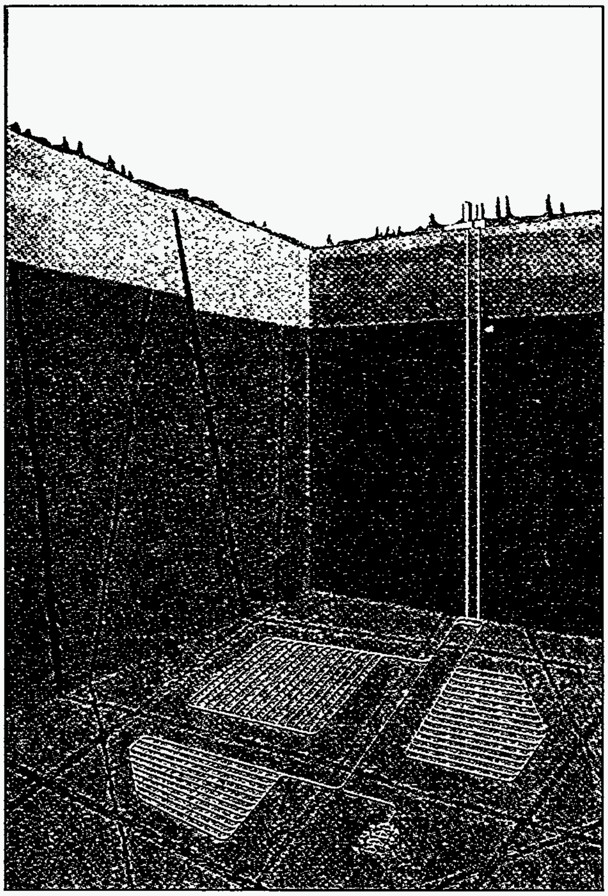

Figure 23.1. Sketch of possible repository layout. The case of 3 HLW emplacement panels on a single level and TRU silos in separate blocks of low-permeability basement between major (layout-determining) water-conducting faults.

Analysis of this concept in the Project Gewähr 1985 study (mentioned above), showed that, for all realistic scenarios analyzed, the performance guideline was met with large margins of safety. In their review of this project, the government concluded that this concept would provide sufficient safety in a crystalline basement having the properties postulated by Nagra. However, only limited data from isolated boreholes were available in 1985, and the Government authorities requested more evidence that suitable rock formations of an appropriate extent could be identified in Switzerland. The government review also strongly recommended that the option

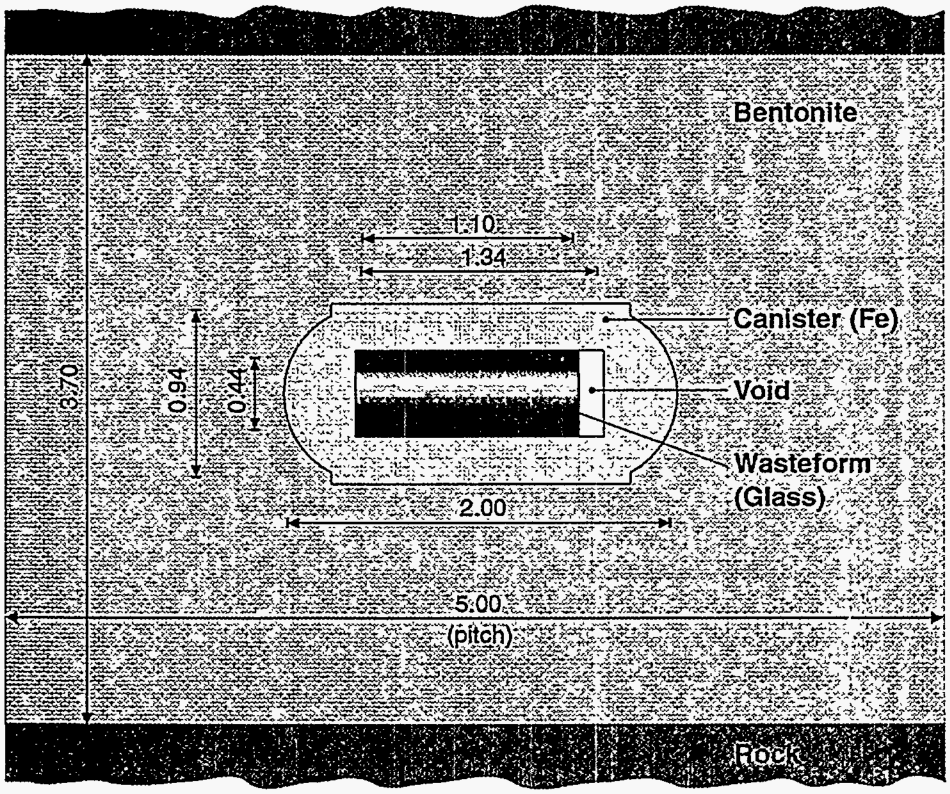

Figure 23.2. Waste emplacement geometry (dimensions in metres).

of disposal in sedimentary formations be considered in more depth. Despite these caveats, the waste disposal issue was no longer tied directly to operation of existing power plants, although it is understood that more evidence of siting feasibility would be required before any applications for new reactor licenses could be sought.

Since 1985, the regional investigation of the crystalline basement has been completed and documented[4]. The geological studies have clearly shown that the extent of the accessible crystalline basement is much less than originally thought due to the presence of a previously unknown, extensive Permocarboniferous trough which cuts the region. Only two restricted areas remain for selection of a possible site, each covering about 50 km^2. Nevertheless, it does seem feasible to find a suitable repository for the required low volume of waste and a strategy for site characterization has been developed. In parallel, investigations of the sedimentary options have proceeded from a desk study to select potential host formations through to identification of specific potential siting areas. The two sedimentary host rocks investigated in detail were Opalinus Clay, which exists in a laterally extensive but rather thin layer in Northern Switzerland, and Lower Freshwater Molasse, where the formations are large but somewhat heterogeneous[5]. For Opalinus Clay, which was identified as the higher priority option for Nagra, a total potential siting area of some 100 km^2 has been identified. Programmes of site-specific studies are now running in parallel in the crystalline basement and the Opalinus Clay.

The next major milestone in the HLW programme will be the demonstration of repository siting feasibility (Project Entsorgungsnachweis) scheduled for 2000. This may include one or both of the potential siting areas studied.

23.4 Ongoing Geological Studies Associated with the HLW Programme

Geological characterization prior to repository construction is planned to progress in three phases. Phase I is a regional study of potential host rocks from the surface, involving seismic studies, investigation from deep boreholes, etc. This phase is followed by more detailed investigations of a potential siting area from the surface (Phase II), and then Phase III underground studies involving construction of an access shaft to the potential repository depth and an underground laboratory.

Phase I has been completed and documented for both the crystalline basement and Opalinus Clay[4,5]. For the former, the most relevant open question to be addressed in Phase II is the distribution of major shear zones within the crystalline basement. A detailed performance assessment (see below) has demonstrated that blocks of low permeability crystalline basement found in the area north of the North-Swiss Permocarboniferous Trough (Fig. 23.3) would be a very suitable host rock. Statistical analysis of major faults identified during Phase I (by geophysics, borehole mapping, mapping surface exposures in the neighboring Black Forest etc.), indicates that the probability of finding sufficiently large blocks for repository construction in this area is high. This statistical model will be tested by drilling a "star" of four inclined boreholes at a site to be chosen in Northern Switzerland (cf. Fig. 23.4). Location of "layout-determining" faults will be on the basis of core-logging, complemented by cross-hole hydro-testing and cross-hole seismic tomography. In addition, a surface campaign of seismics will be carried out. For the crystalline basement, this technique is somewhat limited as it identifies only faults which cause significant displacement in overlying sedimentary formations; no clear determination of structure within the basement is possible using this method.

In contrast, however, the Opalinus Clay formation is a clearly defined seismic reflector which can be well mapped by the planned 3D seismic survey. It is relatively homogeneous in composition and, in the area of interest, shows little evidence of tectonic disturbances. The planned borehole at a site near the village of Benken is aimed primarily to calibrate the seismic studies. Somewhat more problematic in the Opalinus Clay case, however, are the mechanical properties of this rock. Studies to date indicate that emplacement tunnels for HLW and caverns for TRU could be constructed at depths of up to ~800 m, but the extent of tunnel linings required needs to be established based on site-specific, rock mechanics data.

The demonstration of siting feasibility also requires geological input which cannot be obtained from these two sites in any easy way. Some generic data for crystalline and sedimentary rocks can be obtained from other national programmes, but underground laboratories in Switzerland provide a key testing ground for methodology development and use of destructive characterization methods.

Nagra's main underground test site is situated at Grimsel Pass in the Swiss Alps[6]. This facility is situated in granite/granodiorite below ~500 m of overburden. Although tectonically unsuitable for a HLW repository, this site has played an important rôle in the development of the technology required for site characterization. In the present phase of work (Phase IV, 1994-1996), HLW-relevant studies include testing of the limitations of current methods of seismic tomography, examination of the properties of the excavation-disturbed zone around tunnels through crystalline rock, validation of models of radionuclide migration through fractures, and demonstration of the methodology for emplacement of HLW packages. A final Phase V of work at Grimsel, lasting until 2002, is currently being planned.

A further underground test site has been initiated within the scope of an international project, utilizing a road tunnel through the Opalinus Clay at Mt. Terri in the Jura mountains. Studies at this site include examination of water flow paths through this formation and also characterization of the excavation-disturbed zone in this formation.

23.5 MAKING THE SAFETY CASE FOR HLW

Over the last decade or so, several countries have published comprehensive assessments of the safety of various disposal options for vitrified HLW and spent fuel. It has been found that these concepts differ quite significantly from each other and place the emphasis for a demonstration of safety on different parts of the multi-barrier system. At one extreme, the German concept of HLW disposal in a salt dome places emphasis on the role of the host rock in isolating the waste from advective water flow for very long time periods. Another concept with strong emphasis on geological barriers is illustrated by the Belgian concept of disposal in plastic clay. Canadian disposal in granite with very low hydraulic conductivity also emphasizes the geologic barrier, although long-lived container designs are also considered. At the other extreme, the Swedish concept for spent fuel encapsulation in thick copper canisters achieves long-term isolation by depending on the inertness of copper (which gives an expected canister life in the order of a million years and places very modest requirements on the geologic medium). The Swedish concept has also been adapted to Finnish conditions.

The Swiss concept places less emphasis on the retention capabilities of the geosphere or on the performance of individual engineered barriers; rather, it focuses on the behavior of the near field (all engineered barriers in their geological setting) as a whole[7,8]. Following waste emplacement, this near-field environment evolves in a

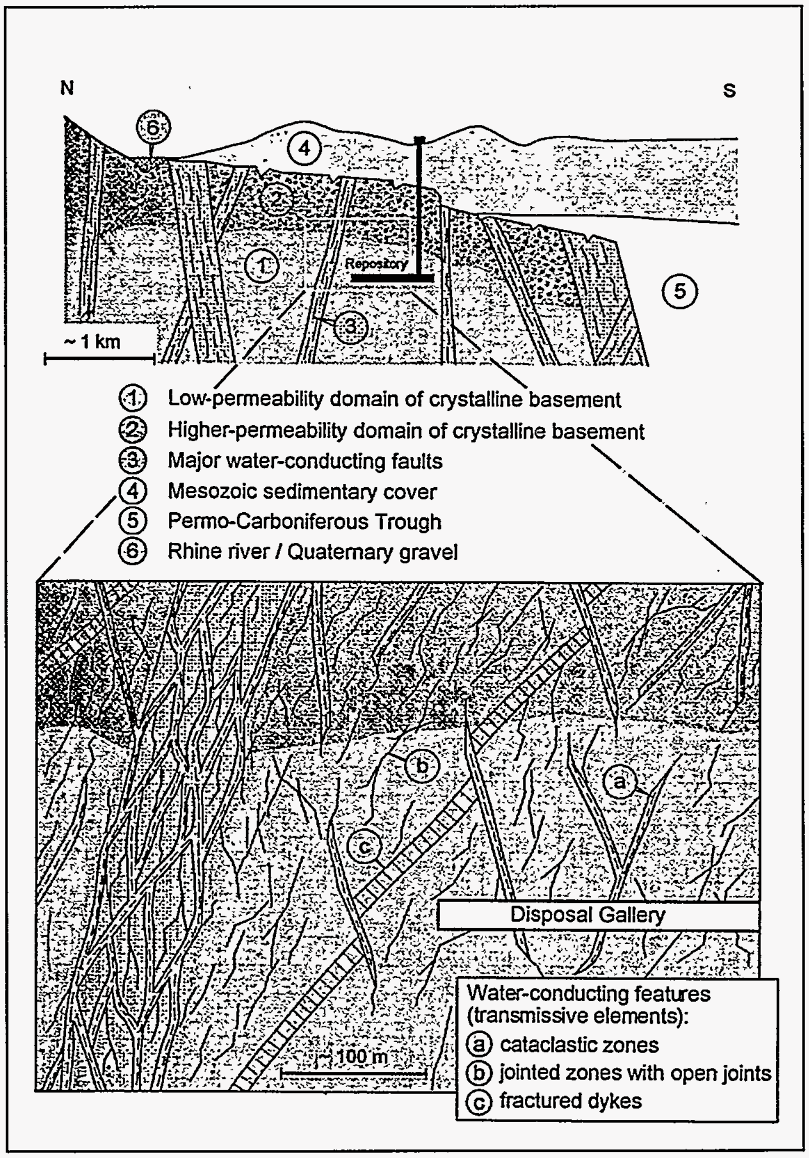

Figure 23.3. Diagrammatic representation of the conceptual model of the crystalline basement.

well defined manner.

After emplacement, water will resaturate the surrounding rock and then invade the bentonite which will swell to seal any gaps. Temperatures in the bentonite will increase due to heat from the canister, but storage of waste prior to emplacement ensures that the maximum temperature in the backfill does not exceed ~160°C. Calculations indicate that complete resaturation may take in the order of hundreds of years. The steel canister corrodes anaerobically at a very low rate (~50 m/year). Only after ~1000 years will mechanical failure occur due to pressure from the expanding bentonite. The water chemistry in the compacted bentonite will be determined by interactions of inflowing granitic groundwater with mineral surfaces in this microporous material. The bulk of the bentonite itself will not undergo any significant mineralogical alteration over relevant timescales (~1 million years).

After canister failure, corrosion of the glass will occur in an environment with effectively stagnant porewater. Corrosion of the glass is expected to gradually release the contained radionuclides over a period of the order of 10^5 years. The release of many radionuclides may, however, be further constrained by their very low solubili-

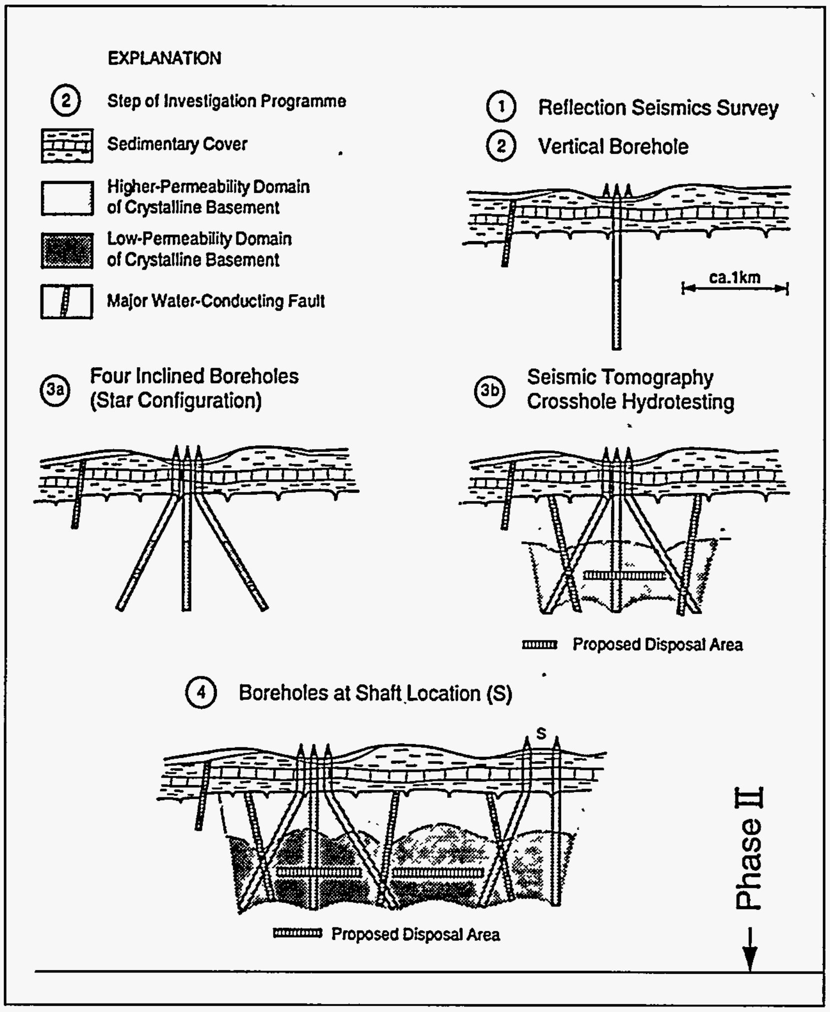

Figure 23.4. Illustration of the investigation concept for the crystalline basement.

ties. Even after radionuclides are released from the glass matrix, output to the geosphere is greatly limited by the transport resistance of the bentonite backfill. Due to its extremely low hydraulic conductivity, solute transport through the saturated bentonite will occur predominantly by diffusion. Sorption processes result in very low diffusivities for many radionuclides, so that their transport time through the backfill exceeds their half-lives, and thus, output is negligible. This analysis of the repository near field is believed to be robust[7], in that it does not take credit for all possible processes which would decrease release rates, and is relatively insensitive to the variation of uncertain parameters within reasonable ranges.

Although sufficient safety can be demonstrated with only minimal requirements on the geology, the geological barrier can also be extremely powerful, reducing the already low concentrations of radionuclides yet further. Under the expected geological conditions in Northern Switzerland, the geological barrier would ensure that, in effect, no releases to the human environment would occur for all timescales which are meaningful (up to one million years). The calculation of retardation of radionuclides in the geosphere is, however, more sensitive to parameter values which are difficult to determine in the field and hence a safety case based strongly on the barrier effect of a fractured geologic medium is less robust.

The demonstration of long-term safety for a HLW repository must be based upon predictive modeling, and it is important to realize the capabilities and the limitations of such models. The mathematical models used in performance assessment are supported by a range of laboratory and field experimental studies, but the extrapolation of such work to very long timescales must also be justified. Enhanced confidence in our understanding of long-term performance of the near-field, in particular, may be illustrated by using natural analogues.

A natural analogue is a process that has occurred in the past and is similar to those that are predicted to occur in the future evolution of a repository. The proposed corrosion rates of the waste matrix and canister in the expected chemically reducing environment can be supported by observation of the preservation of archaeological glass and steel artifacts over millennia. Mineralogical stability of bentonite can be shown on an even longer timescale by observations of natural bentonites that have remained unaltered for millions of years in conditions comparable to those expected in the repository. Even the ability of clay to isolate radioactive substances can be illustrated by natural ore bodies in appropriate geologic settings.

A safety case made on the robustness of a system of engineered barriers appears to be appropriate to the Swiss geological environment, and it is interesting to note that a very similar concept has been adopted by Japan (a country with even more complex and tectonically active geology than Switzerland) in their H-3 performance assessment.

23.6 The Swiss Programme in an International Context

The Swiss waste management programme, although relatively small in terms of budget and manpower, is very wide in scope, with one site currently being characterized in detail for a L/ILW repository and two types of host rocks under investigation for disposal of vitrified HLW and long-lived ILW. This programme is only feasible if priorities are set and adhered to, and if maximum advantage is taken of work performed elsewhere. Therefore, extensive use is made of international collaboration agreements in order to spread the work load. Individual information exchange agreements with other programmes have allowed effort to focus in specific areas. For example, Switzerland could deliberately concentrate on studies of steel canisters for HLW because Sweden has concentrated on copper, and a bilateral Nagra/SKB agreement provided for exchange of results. Such agreements also allowed direct cooperation/cofunding of larger studies, such as the Japan/Sweden/ Switzerland (JSS) study of the leaching of vitrified HLW. Cooperation with the US DOE has allowed results from the Swiss underground test site to be made available to modeling groups in the US who, in turn, make their interpretations available to Nagra.

Apart from active participation in the IAEA and the NEA, Nagra has formal agreements with the European Economic Community (EEC), United States (DOE, NRC), Sweden (SKB), Finland (Posiva), France (CEA and ANDRA), Belgium (ONDRAF, CEN/SCK), Germany (GSF/BRG), Japan (PNC), Spain (ENRESA), Taiwan (AEC) and the United Kingdom (NIREX). Informal collaborations extend the list further.

23.7 Conclusion

Despite its small size and limited nuclear power capacity, Switzerland has succeeded in establishing an internationally recognized programme for management of radioactive wastes. The restricted size makes lines of communication shorter and coordination of effort simpler. The relatively strong economy makes the financing of projects, without the benefits of scale, a feasible proposition, although the economic sense of establishing various small-scale projects through the world can be questioned. Sound technical projects can be developed and implemented with limited human resources, provided that care is taken to make polyvalent use of expertise and to profit from mutually beneficial collaboration with other national programs. Even front-line science and advanced engineering skills are, however, of little use if public opposition prevents their application. Hence, it is important that a waste management organization like Nagra devotes strong efforts to communication with the public at all levels.

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CHAPTER 24

LOW LEVEL RADIOACTIVE WASTE MANAGEMENT IN TAIWAN

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24.1 BACKGROUND

The commercial operation of Chinshan Nuclear Power Plant (NPP) Unit one marked the beginning of Taiwan's nuclear power program. There are now three NPPs each consisting of two units, in operation. With a generating capacity of 5,144 MWe, nuclear power produces some 30 percent of the electricity supplies in Taiwan. However, the nuclear power component is decreasing as the power demand increases. In order to meet the increased power demand, the Taiwan Power Company (TPC) , a state-run and sole electricity utility in Taiwan, decided to build one more nuclear power plant with two reactors at the Yenliao Site in addition to the other sources. Detailed information about Taiwan's nuclear power program is shown in Table 24.1.

As far as low level radwaste (LLRW) is concerned, TPC is the principal source contributing more than 90 percent of total volume of waste produced in Taiwan. Small producers in the medical and research institutes and universities are responsible for the remaining 10 percent.

24.2 RADWASTE MANAGEMENT POLICY AND ORGANIZATIONAL SCHEME

24.2.1 Policy

On the 16th of September 1988, the Executive Yuan (the Cabinet) promulgated the Radwaste Management Policy (RWMP) that set up the principal guidelines to enable the Taiwan nuclear industry to plan and manage its radwaste. Highlights of the RWMP concerning LLRW are summarized as follows:

- the radwaste producers should strive to minimize the waste generation rate and reduce the volume;
- the responsibility of safely treating, transporting, storing, and disposing of radwaste should rest with the producer. Therefore, the producer is responsible for the necessary expenses; and
- an LLRW disposal site should be located by 1996, and operational by 2002.

24.2.2 Organizational Scheme

The organizations related to radwaste management are shown in Figure 24.1. Both the Atomic Energy Council (AEC) and the Ministry of Economic Affairs (MOEA) are under the Executive Yuan. The Fuel Cycle and Materials Administration (FCMA), a subordinate orga-

Table 24.1. Information on nuclear power plants in Taiwan.

Unit	Reactor Type	Installed Capacity	Commercial Operation	Status
Chinshan 1 (C1)	BWR/4	636	1978	operating
Chinshan 2 (C2	BWR/4	636	1979	operating
Kuosheng 1 (K1)	BWR/6	985	1981	operating
Kuosheng 2 (K2)	BWR/6	985	1982	operating
Maanshan 1 (M1)	PWR	951	1984	operating
Maanshan 2 (M2)	PWR	951	1985	operating
Yenliao	ABWR	1300	2000 (scheduled)	bidding

nization to the AEC, assumes regulatory control over radwaste management matters. The Institute of Nuclear Energy Research (INER) was empowered by AEC to take responsibility for collecting radwaste generated by small producers and treat the waste as necessary. In TPC, the Nuclear Backend Management Department (NMBD) and the Nuclear Operation Department (NOD) take care of radwaste generated by the NPPs. NOD's major responsibility is to supervise treatment and storage of LLRW within the NPPs, whereas NMBD is responsible for radwaste transportation, the operations of both the Lan-yu storage site and the Volume Reduction Center, but more importantly, the final disposal of LLRW in Taiwan.

24.3 Technical aspects of LLRW management

Before introducing the detailed technical issues of LLRW management in Taiwan, it is better to review the LLRW management diagram (see Fig. 24.2).

24.3.1 Radwaste Generation

LLRW in Taiwan can be divided into two categories: wet waste and dry active waste. Wet wastes, namely: evaporation residues, filter sludges, and spent bead resins, are first solidified in carbon-steel drums and then stored in structurally safe warehouses. Dry active wastes, which are mainly waste paper, clothes, plastics, wood materials, metal, etc., are either segmented or shredded and also stored in warehouses. The cumulative amounts of radwaste generated through August 1994 are listed in Table 24.2. Cement is the most commonly used solidification agent for wet waste. However, bitumen is used in solidifying incinerator ash.

Thanks to waste reduction efforts implemented by industry, the annual radwaste generation rate at the three nuclear power plants has been decreased from more than 12,000 drums prior to 1990 to less than 8,000 drums afterwards. A particularly significant reduction has been achieved for solidified wastes. Together with radwaste generated by small producers, the present annual radwaste generation rate is approximately between 5500 and 6500 drums. Up to now, almost half of the radwaste drums have been shipped to the Lan-Yu National Storage Site for extended storage. However, the remainder of the radwaste is stored in warehouses on site. As the nuclear facilities are nearly running out of storage capacity, a computerized and improved and better equipped on-site warehouse at two of the three NPPs and at INER is either being constructed or is planned. These new facilities are scheduled to commence operation in the near future.

24.3.2 Waste Volume Reduction

Reducing the volume of both combustible and compactable wastes is justified as a good way of mitigating storage pressures given the ever-increasing quantities of

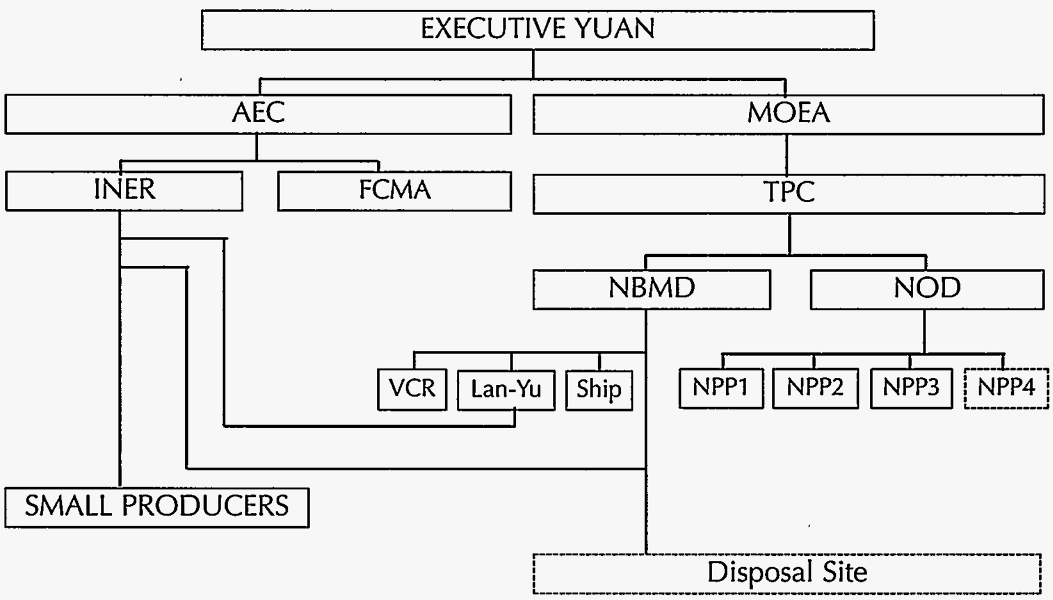

Figure 24.1. Organizations related to radwaste management in Taiwan.

Table 24.2. Total amount of LLRW in Taiwan (unit: 55 gal. drums).

Plant	Solidified Waste	Non-Solidified Waste	Totals
NPP I (C1&C2)	43,684	31,474	75,158
NPP 2 (Kl&K2)	53,090	20,876	73,966
NPP 3 (Ml&M2)	7,004	3,627	10,631
INER	13,774	328	14,102
Total	117,552	56,305	173,857

LLRW. Hence, TPC built a Volume Reduction Center at Kuosheng NPP site. The center comprises a controlled air incinerator and a supercompactor. With a capacity for burning 100 kg/hr of combustible waste and compressing five waste drums per hour of compactable waste, this center is able to eliminate about 3,500 waste drums annually. This has helped to relieve storage problems to a great extent. The important operating parameters of the Volume Reduction Center are shown in Table 24.3. INER has also constructed a controlled-air type of incinerator with a burning rate of 40 kg/hr to treat combustible wastes originating from small producers islandwide.

24.3.3 Lan-Yu Interim Storage

The National Lan-Yu Storage Site provides off-site interim storage for solidified radwaste. This site is located on the small island of Lan-Yu that has an area of about 45 km^2, and indeed, was originally designed as a port of departure for sea dumping that is no longer allowed. Twenty-three semi-underground engineering trenches were constructed on the site, providing a storage capacity of 98,000 drums with three drums being stacked vertically. As of August 1996, the site had received about 97,700 waste drums, and it is anticipated to be full by 1996. However, in a continual search for sufficient storage space and to allow ample time for proceeding with the LLRW disposal program, TPC, the site owner, plans to expand the storage capability by adding six better shielded trenches with a capacity of 59,000 drums. Simultaneously, an optimized waste drum loading pattern will be adopted, making better use of the land. The environmental impact report for this project is under review by the AEC's Environment Evaluation Committee.

24.3.4 LLRW Transportation

Due to the need to continually ship solidified LLRW to the Lan-Yu site before the disposal facility was commissioned, TPC built a modern LLRW transport ship, Teen-Kung No. 1, to replace an old ship in 1991. The new ship is 53 meters long and has a deadweight of 737 metric tons at the designed draft. It can reach a speed of 11.5 knots. Furthermore, it features a double-shell hull, auto-

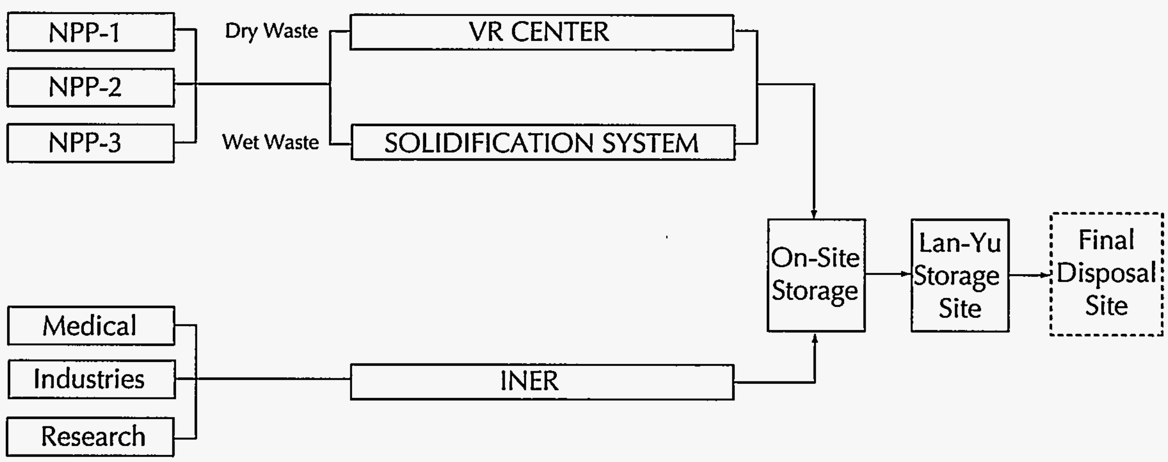

Figure 24.2. Diagram of low level radwaste management in Taiwan.

Table 24.3. Important operating parameters of the Volume Reduction Center.

Parameters	Value
Incinerator:	
Burning rate (kg/hr)	100
Operating temperature (°C)	
1st chamber	700 - 900
2nd chamber	1000 - 1200
Volume reduction ratio	30-100:1
Weight reduction ratio	30-40:1
Supercompactor:	
Compacting force (ton)	1500
Feeding rate (drums/hr)	5
Volume reduction ratio	3 - 5

matic navigation, satellite-relayed communication, and state-of-the-art radiological protection equipment. It can carry up to 576 waste drums per shipment.

24.4 LLRW FINAL DISPOSAL

Low level wastes presently stored on site or in the Lan-Yu site have to be permanently disposed of in a safe manner. Due to the RWMP direction and in light of the fact that TPC contributes 90 percent of the LLRW generated in Taiwan, TPC has been designated to assume this work.

24.4.1 Regulatory Requirements

According to the "Low Level Radwaste Land Disposal Licensing Regulations" issued by AEC-RWA, the annual dose to any member of the public resulting from release of radioactivity from a disposal site must not exceed 25 millirems (0.25 mSv). When the individual dose is less than 1 millirem (0.01 mSv) and the collective dose less than 100 man-rem (1 man-Sv), the disposal site can then be freed from institutional control. The regulations also point out a set of siting requirements for the final disposal program. They are that the site should:

- be situated in an area with low population density and low development;
- avoid an area in which tectonic activity, geological processes, hydrological and geohydrological conditions could endanger the safety of the disposal facility; and
- be kept away from an area where geological and hydrological data are too complicated to be adequately evaluated.

24.4.2 Geological conditions in Taiwan

Taiwan measures about 36,000 km^2 in area with a spindle shape for the island. There are 81 islets spreading out in the surrounding Pacific ocean, and 64 of them are known as the Penghu Island Group, or the Pescadores, in the Taiwan Strait (Fig. 24.3).

Located at the boundaries between the Eurasian plate and the Philippine sea plate, Taiwan island reaches a maximum elevation of about 4000 m as a result of the compression and shear forces. It is an arcuate island extending its shorter arm eastward to the Ryukyus and its longer arm southward to the Philippines. The backbone of this mountainous island is the Central Range which is mainly Tertiary in age. It is fringed on the west by the Foothill Zone and separated on the east from the Coastal plain with the very shallow Taiwan Strait farther west; east of the Coastal Range is the deep Pacific Ocean. The offshore islets of Taiwan include the Penghu Group in the Strait and Liitao and Lan-yu off the southeast coast. Kinmen and Matsu are two islands close to mainland China covered with Mesozoic granitic gneiss which may be a surface extension to Taiwan. In the less tightly compressed northeastern and southwestern parts of the mountain complex of the Central Range and Foothills of Taiwan, there is the Ilan plain and the Pintung Valley, each in the form of an intramundane trough intruding from the sea into the island.

24.4.3 LLRW Final Disposal Program

The TPC's program plan for LLRW disposal will be carried out in the following 6 phases:

Phase 1. Selection of Disposal Site and Method
The site selection criteria and process were developed taking into account Taiwan's local conditions and foreign experience. Based on the available geological and socio-environmental situation, a handful of candidate sites will be identified in accordance with siting criteria. Further investigations, including core drilling and laboratory testing on those candidate sites will then follow. Various land disposal methods will be assessed against each candidate site condition to determine those that are suitable. In this manner, the most favorable disposal site

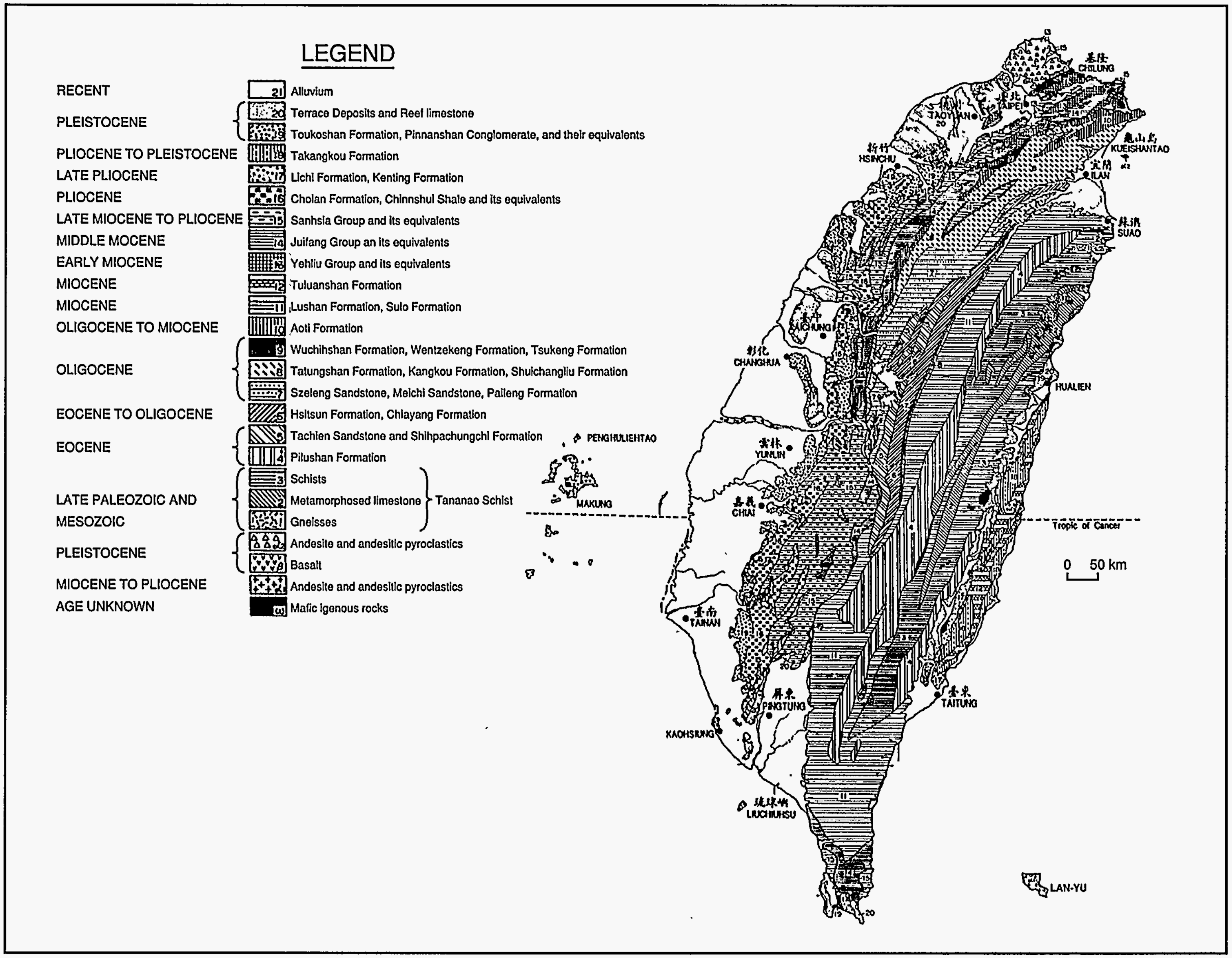

Figure 24.3. General geologic map of Taiwan.

and method can then be selected.

Phase 2. Environmental Survey and Assessment
The environmental data survey and documentation will be conducted in parallel with drilling investigations on the above mentioned candidate sites. Results of the environmental assessment will become part of the attributes used for evaluating and comparing candidate sites.

Phase 3. Site Characterization, Engineering Design and Licensing
It is expected to take at least two years to complete this phase. During the phase, the site characterization, engineering design and the detailed safety analyses will be undertaken to support the presentation of a construction license application.

Phase 4. Site Construction
Depending on the site condition and disposal method, it is expected to take three years to complete the initial phase in the construction of the disposal facility. An operating license application is scheduled to be submitted to the government in 2001 for review.

Phase 5. Operation
The disposal facility is programmed to be commissioned in early 2002 if everything goes as planned.

Phase 6. Post-Operation Monitoring
After the disposal facility ceases operation, it will be backfilled, stabilized, and covered with earth and vegetation. The disposal site and its vicinity will then be monitored until the radioactivity in the disposed waste has decayed substantially and no longer presents a risk to the environment.

The milestones for each phase of the disposal program are shown in Table 24.4. However, due to nontechnical factors, Phase 1 has been postponed to the end of 1996.

24.5 PUBLIC ACCEPTANCE

The importance of securing public acceptance in proceeding with the LLRW management program has long been recognized by the nuclear industry. The continuing receipt of protests against the storage of LLRW in the Lan-Yu Storage Site from native residents is one of the examples of this kind. Another example could be justified by the strong and violent protests from an opposition party in the parliament to freeze the budget for the construction of the fourth NPP. Currently, the opposition party has about one-third of the seats in the parliament. As elsewhere in the world, nuclear safety and radwaste management in Taiwan have become the two major issues of the anti-nuclear movement.

It is anticipated that, in the future, the establishment of a LLRW final disposal facility could receive many objections from the public since the disposal site will be situated at a given location for a few hundred years. The radwaste people in the nuclear industry are deliberating on how to get the public involved at an early stage in

Table 24.4. Taipower's overall program plan for LLW final disposal.

Plan / Phase	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002
Phase I: Selection of Candidate Site and Disposal Technology (10/92-9/95)											
Phase II: Environmental Impact Assessment (5/93-9/95)											
Review by Government Authorities (10/94-9/95)											
Phase III: Site Characterization and Engineering Design (10/95-3/99)											
Review by Government Authorities (4/98-3/99)											
Phase IV: Site Construction (4/99-9/2002)											
Review by Government Authorities (4/2002-9/2002)											
Phase V: Facility Operation (10/2002-)											

proceeding with any of the LLRW management programs. To clearly separate the issue of radwaste from that of nuclear power plant development may be strategically important in resolving the radwaste issue. Nevertheless, both to ensure the safety of the final disposal site and to provide a satisfactory financial aid to offset local objections may be the first two essential tasks to be worked on among other things.

24.6 CONCLUSIONS AND RECOMMENDATION

Taiwan is a country of scare natural resources of energy, and, therefore, the use of nuclear energy becomes a necessity. The management of radwaste arising from the use of nuclear power has to be safely planned and implemented. To locate a site, as early as possible, to permanently accommodate LLRW in Taiwan is considered the top priority among other management activities. Since the country is heavily populated and small in area, it welcomes any form of regional cooperation in the disposal of radwaste. Indeed, international cooperation in radwaste disposal is believed to be of benefit to the whole world.

It is hoped that an active program of regional cooperation on the disposal of LLRW can be initiated by a competent organization, such as PBNC (Pacific Basin Nuclear Conference), in light of the potential benefits to this region.

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CHAPTER 25

PROGRAMME AND RESULTS OF INITIAL PHASE OF RADIOACTIVE WASTE ISOLATION IN GEOLOGICAL FORMATIONS IN UKRAINE

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Abstract: The concept and a programme for radioactive waste disposal in geological formations of Ukraine have been developed. On the basis of certain criteria, an evaluation of the territory of Ukraine has led to the selection of three geological regions and three types of formations that are favourable for RAW disposal. The programme of research and development includes three stages: preparatory (1993-95), preparatory/experimental (1995 -2004), and preparation for construction (2005-2010). The completion of the preparatory stage resulted in the selection of zones and a number of candidate sites that are favourable for RAW isolation.

25.1 INTRODUCTION

Ukraine has been forced to develop research and development (R&D) programmes on radioactive waste (RAW) management due to their accumulation in significant quantities. This is a result of the rapid development of nuclear power and other RAW-producing industries, as well as the consequences of the Chernobyl disaster. According to the generally accepted point of view, a realistic solution for the RAW disposal problem is isolation in geological formations. The importance of this problem resulted in a research program initiated by the State Committee on Nuclear Power Utilization (Goskomatom), National Academy of Sciences, State Committee on Geology, and some other organizations.

The preparatory stage of research on RAW isolation in geological formations has been completed. Ukrainian scientists have developed the concept and a programme of R&D (experimental and methodological studies on a pilot scale as applied to geological and mining activities). The territory of Ukraine has been assessed as to the conditions for RAW isolation, and geological regions and formations favourable for this purpose have been selected. Regional studies, to be discussed below, have resulted in the selection of a number of favourable zones (areas) within these regions, and candidate sites have been selected. Simultaneously, preliminary analyses of the main engineering and construction problems related to RAW isolation have been made.

However, Ukraine is still lagging considerably behind in the field of R&D as compared to the countries that have been developing their programmes over several decades. Ukraine has established scientific relations with specialists in the field of RAW management. As a result of an international conference on "Isolation of RAW in Geological Formations" that was held September 20-24, 1994 in Kiev, a basis for cooperation in this field has been initiated with eastern European countries.

Investigations on R&D have been sponsored by Goskomatom partly by finances from budgets of participating institutions, as well as by special funds from the State Committee on Science and Technology for individual projects. These investigations are carried out by specialized multidisciplinary research teams, from 23 R&D institutions (Institute of Geological Sciences, State Committee on Geology, Kiev University, Goskomatom, etc.).

25.2 DESCRIPTION OF THE WORK

25.2.1 General Concept

The general concept for RAW isolation in geological formations in Ukraine is based on the experience of advanced countries, IAEA basic principles and technical criteria adapted to geological, socio-economic and ecological conditions in the Ukraine[1-5]. The principle of long-term (over 10,000 years) RAW isolation is based on the idea of disposal as a geological engineering system that must satisfy a range of conditions (final form of RAW, disposal in deep geological formations at an appreciable depth, special engineering barriers, etc.).

In the preparatory phase of the R&D programme, certain criteria were adopted, and an evaluation of the territory of Ukraine was carried out[3]. As a result, three geological regions (Fig. 25.1) and three types of geological formations, favourable for RAW disposal, have been selected (see below).

The amount of waste to be isolated is (metric tons): spent fuel - 27,000, decommissioned waste - 12,000, Chernobyl zone - 20,500, for a total of about 60,000. The future accumulations of spent fuel are estimated to be (metric tons): year 2000 - 2020, year 2005 - 3725, year 2010 - 5460.

After an appropriate period of cooling, or reprocessing procedure, the spent fuel has to be encapsulated. Technological waste must be conditioned and solidified. As for the Chernobyl zone wastes, there are two variants: straight burial and separation (enrichment) for volume decrease. Two types of canisters have been considered: stainless steel and steel-copper. After being reprocessed in Russia, the spent fuel has to be returned in standard containers intended for burial.

In selecting procedures for repository construction, world experience in underground methods, and current projections were taken into consideration. Our approach is to develop procedures that are the most simple and least expensive. The repository will consist of a wide transport tunnel and system of galleries for disposal (Fig. 25.2). Disposition of the different types of RAW involves: spent fuel in short boreholes in the floor of gal-

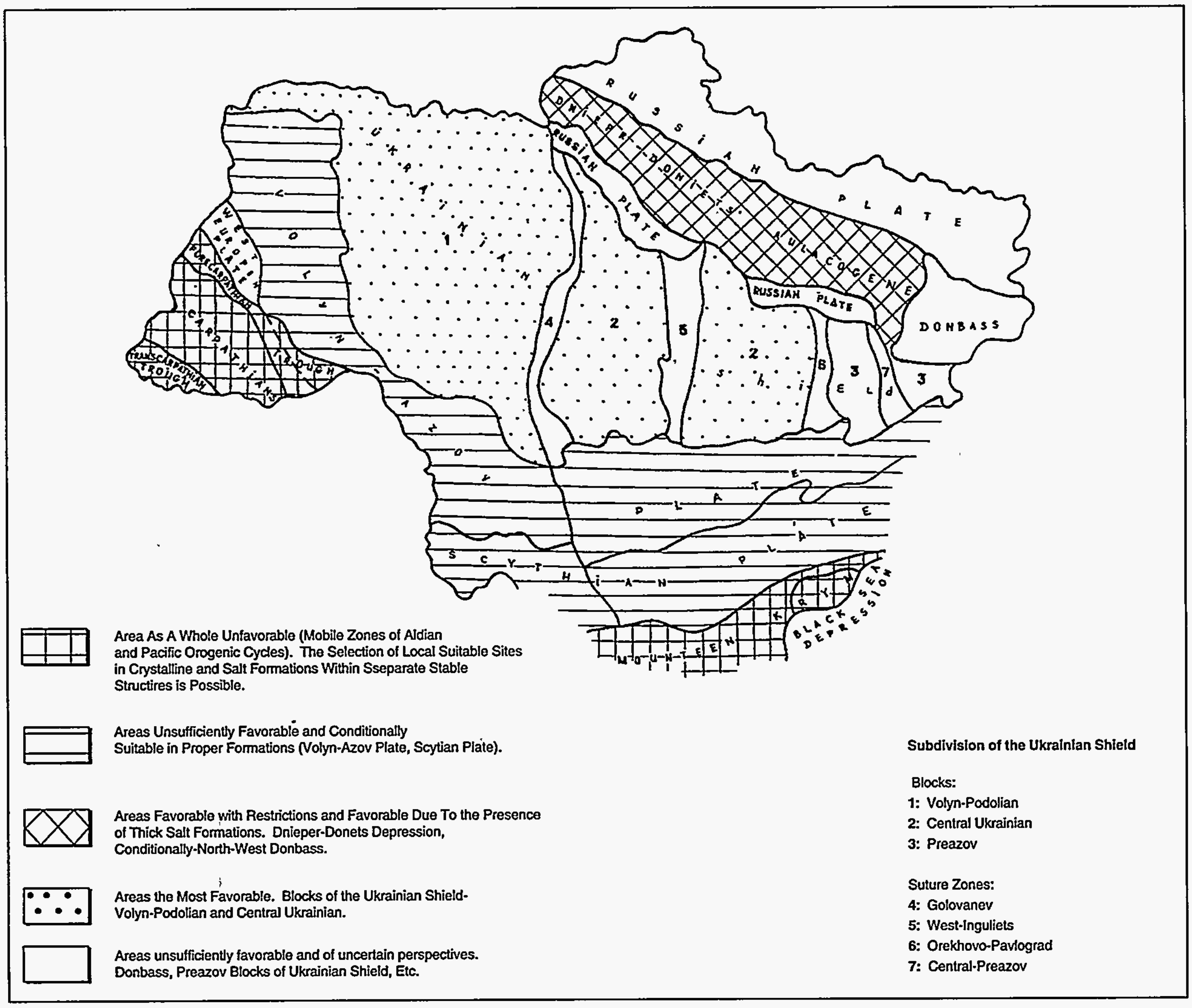

Figure 25.1. Subdivision of the Ukraine on conditions of RAW isolation in geological formations.

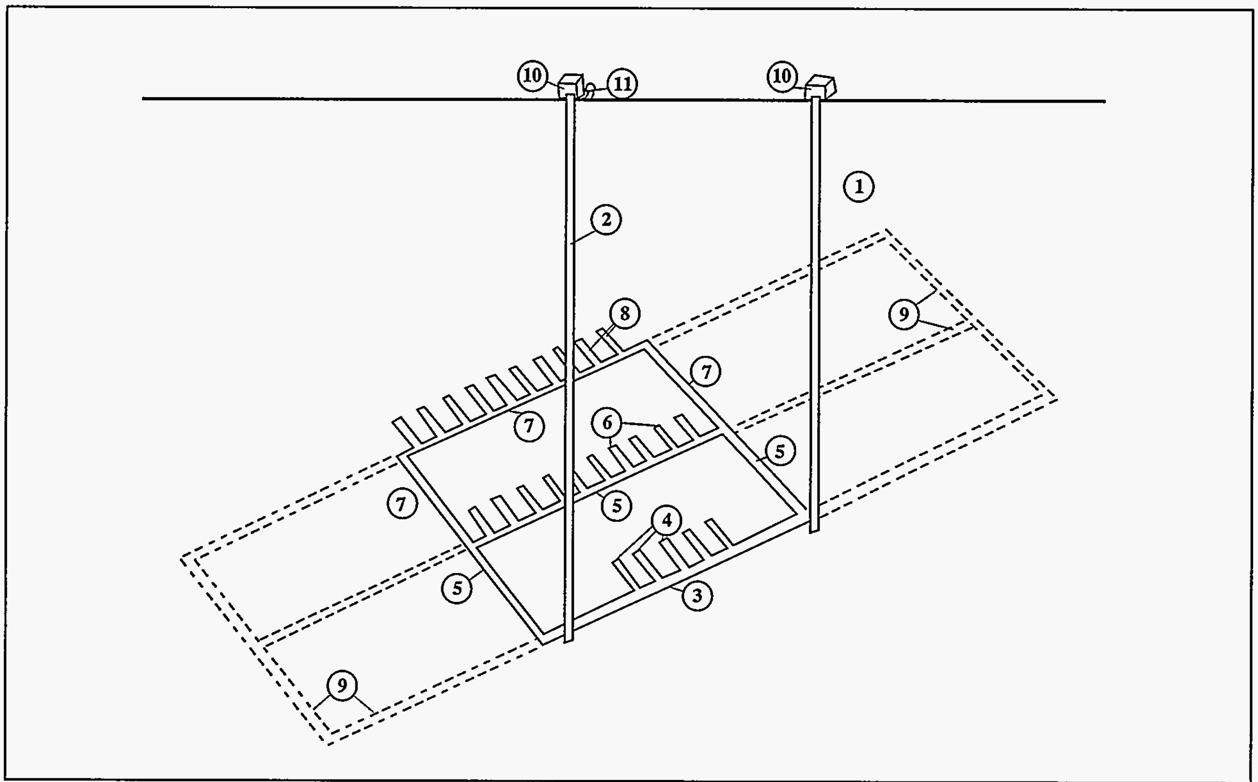

Figure 25.2. Concept of underground experimental laboratory and storage. Legend: 1 - Mine shaft; 2 - ventilation shaft; 3 - crosscut of laboratory; 4 - chambers of laboratory; 5 - crosscut of the first stage of storage; 6 - chambers of the first stage of storage; 7 - crosscut of the second stage of storage; 8 - chambers of the second stage of storage; 9 - crosscut of the following stages ; 10 - mine surface building; 11 - fan installation.

leries, and technological waste and Chernobyl waste in special cells in the galleries. The system of engineering barriers includes the matrix, a buffer, containers, compactors, and backfilling material of a bentonite composition. Sometimes, a protective covering is needed on the cavern walls. Special non-blasting methods of excavation for maximum preservation of rock integrity have to be used.

The construction of an underground research laboratory (URL) is planned as the first stage in developing the repository. The investigations in the URL are of a traditional nature but the programme may be shortened using results from actual world experience.

25.2.2 Research and Development Programme

The purpose of the R&D programme is to develop the complex measures needed for RAW isolation (long-term storage and final burial) in geological formations. The programme is based on total safety for the population and environment using principles elaborated by IAEA. The overall programme for R&D includes: site selection and investigation, projections, exploration, construction, testing, exploitation and final closure of the isolation facility. There are seven topical areas: (1) mining/geology (including geological exploration); (2) technology; (3) social; (4) regulatory; (5) legal; (6) management and (7) construction.

The area of mining/geology is actually central and is relatively independent due to its long term duration and the essential value of the data being collected. The main tasks in the mining/geology area are as follows:

1. Conducting theoretical investigations of geological, geochemical, hydrogeological, geomechanical, mining, thermophysical and other problems connected with site selection, exploration, construction, exploitation and closure of the isolation facility, as well as safety and the development of a methodology of investigations.

2. Evaluation of the territory of Ukraine from the point of view of RAW disposal.
3. Regional studies to elaborate on selection criteria, structures and the selection and evaluation of sites.
4. Supervision of exploration on selected structures and sites.
5. Construction of URL to carry out selected experiments.
6. Perfection of construction and technological parameters for the RAW isolation facility based on the synthesis of exploration data, experiments in the URL etc.
7. Develop a prognosis for the functioning of the isolation facility under the influence of the effects of geological evolution and scenaria of possible catastrophic events (safety analysis).
8. Develop a monitoring system and system of control (management).
9. Provide a basis for controlling construction of the RAW isolation facility.
10. Provide a basis for supervising the exploitation and closure of the facilities.

The programme of R&D includes the following stages:

1. Preparatory - 1993-95
   Goal: elaboration of concepts, selection of sites, exploration;
2. Preparatory/experimental - 1996-2004
   Goal: exploration, construction of URL, collection of experimental data during construction and exploitation of the isolation facility, and
3. Preparation for construction - 2005-2010
   Goal: final preparation for construction of RAW isolation facility (eventually the beginning of the construction).

The durations of these stages are not yet firm and will actually depend on the financial situation.

25.2.3 Methodology of Scientific Investigations.

The generally accepted concept of the repository as a multibarrier, geological/engineering system takes into account that the rock formation, as a main barrier, is the leading factor in determining the safety of long term isolation. That is why a comprehensive investigation of the geological environment provides a foundation for investigations in the area of mining/geology.

The final goal in the preparatory stage of the R&D programme is the selection of site(s) for the isolation of RAW. This goal can be achieved by solving the following tasks in a proper hierarchical sequence that is correlated with the stages of programme investigations mentioned above:

1. Evaluation of the territory of Ukraine from the point of view of RAW isolation;
2. Selection of geological regions and formations potentially favourable for RAW isolation;
3. Regional analysis of potentially favourable formations in a hierarchical sequence. Region-Zone (group of structures)-Local Structure (site), (Fig. 25.3); and
4. Selection and evaluation of sites.

Tasks 1-3 and part of 4 have already been accomplished.

As a result of the evaluation and ranking of 12 geological regions in Ukraine, only three have been selected as favourable for RAW disposal: (1) Ukrainian shield; (2) Dnieper-Donets depression; and, (3) northwestern Donbass. Conditionally, the southwestern slope of the east European platform of the ancient Volyn-Azov plate is under investigation.

The selection of formations in these regions was made on the basis of an initial evaluation of parameters and by analogy with world experience. By mean of this approach, three types of potentially favourable formations have been selected: crystalline, salt and argilaceous. The next step was to carry out a regional analysis of formations.

The hierarchical sequence of investigations (Regional-Zonal-Local) is similar for all formations. The methodology of the selection process on these three levels is based on the usage of a set of mining/geology area models, categorized on different scales according to the level of investigation.

The results of this selection of a set of models (as well as data from social, economic, ecological and other studies) provide the basis for setting up criteria for the selection, comparison, estimation and ranking on zonal and local levels. This set includes three groups of practically equal importance:

I. Safety (technical-geological)
   a. tectonic
   b. neotectonic
   c. seismic
   d. hydrogeologic
   e. type of formation

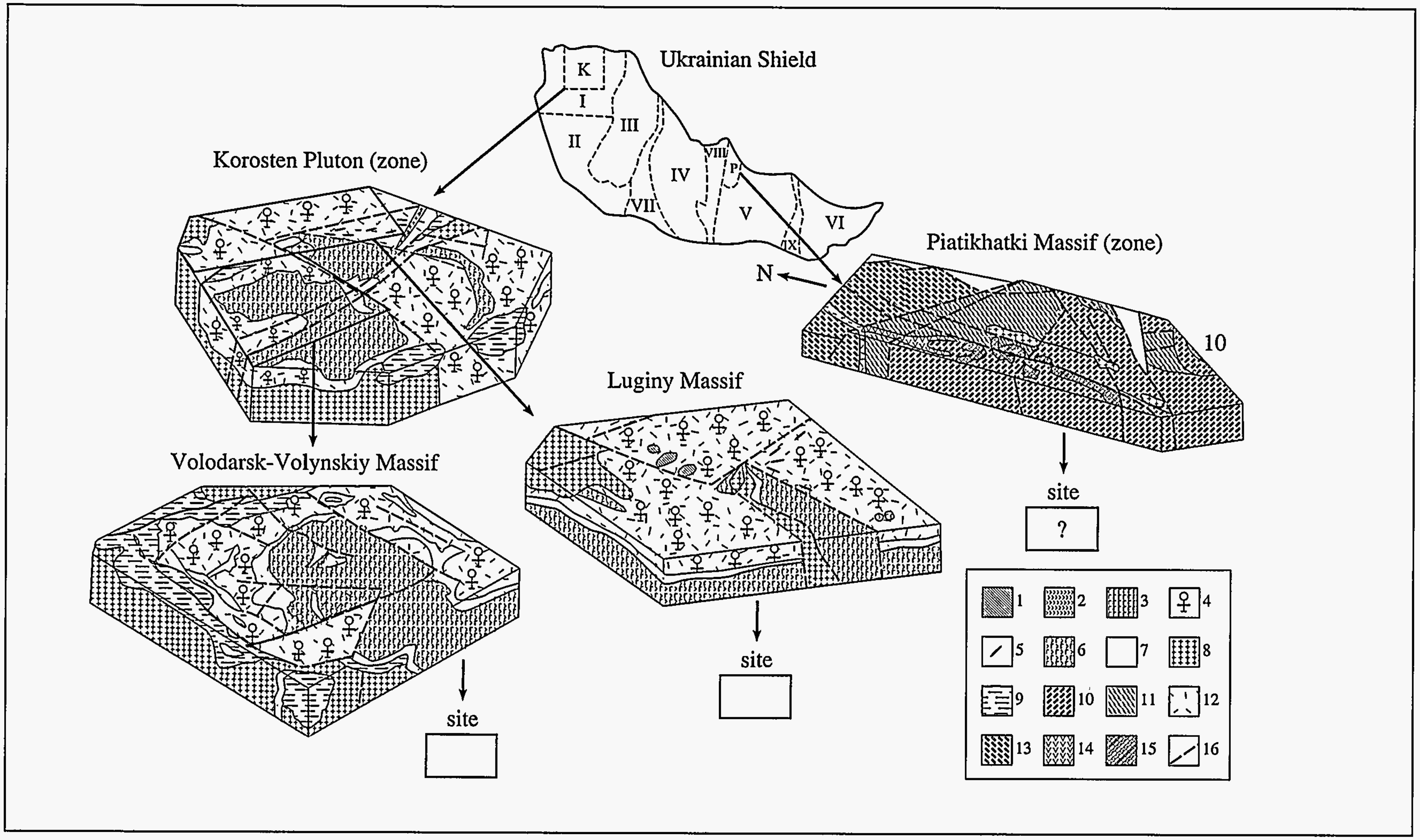

Figure 25.3. Principles of site selection in Ukrainian shield: I - Volyn, II - Podolia, III - Bielaya Cerkva, IV - Kirovograd, V - Near-Dnieper, VI - Near-Azov Pavlograd, VII - West Ingoulets, VIII - Golovanev, IX - Orechovo. Legend: 1 - alkaline syenites; 2 - monzonites, gabbro monzonites: 3 - olivine gobbros, rarely gabbro-peridotites, gabbro-pyroxenites; 4 - granite-porphyries, rapakivi and rapakivi-like granites; 5 - dikes of diabase and gabbro-diababase; 6 - anorthosites, gabbro-anorthosites; 7 - norites, gabbro-norites; 8 - granites (Zhitomir and Kirovograd types); 9 - biotite gneiss, quartzitic sandstones; 10 - plagiomigmatites, plagiogranites; 11 - granites, migmatites; 12 - biotite granites, aplitic and pegmatoid granites; 13 - tonalites, plagiogranites, plagiomigmatites; 14 - andesite and diabase metaporphyry amphibolites; 15 - graphitic and biotites shales, metasandstones; 16 - faults.

f. goemechanic
g. geochemistry (including waste composition)
h. geomorphologic
i. hydrologic
j. climatic
k. technogenic/ecologic conditions
l. mineral deposits

II. Social-political
a. demographic
b. psychologic
c. contamination from Chernobyl

III. Technologic complexity
a. construction cost
b. technologic complexity

On the zonal and local levels, the results of the social-economic and ecological investigations must be developed in different degrees of detail.

In the areas of mining/geology, the set of models that are being used in the hierarchical sequence of regional investigations includes both static and dynamic aspects. The dynamic aspect must consider two variants: evolutionary and revolutionary (catastrophic, or maximum project risk). Every model has its own tasks, objects and phenomena for investigation, but all models are integrated within the whole set. An understanding of the functioning of the disposal facility has to be developed from an appropriate synthesis of these models.

These models are developed using three kinds of data: theoretical, computational (mathematical, statistical, probabilistic) and experimental. The experimental data are obtained as a result of URL investigations. The dimensions of these models are generally known: near field and far field. The objectives of the investigations for this work are well described in the literature. The main task in analyzing the functioning of a disposal facility is the prognosis of its long term safety. This prognosis has to be developed within a framework that includes scenaria of evolutionary and catastrophic phenomena. One of the terminal tasks of modelling is the estimation of radionuclide behaviour in the biosphere (accumulations in surface waters, sorption by clays and organic matter etc.). The monitoring of an RAW disposal facility may be realized in the near field by means of direct observations *in* situ; in far field, by mean of special boreholes and surface observations.

25.3 RESULTS OF REGIONAL STUDIES

Regional studies have been carried out in the Ukrainian Shield, Dnieper-Donets Depression and Donbass.

In the Ukrainian shield (Fig. 25.3), two zones have been selected as favourable for RAW disposal: (1) Korosten pluton and a group of structures in the middle of the Near-Dnieper area, where the preferable type of rocks, granites and gabbro-anorthosites of Proterozoic age are found; and (2) in salt domes of the Dnieper-Donets depression in the northeastern, and southwestern marginal zones, and in the southeastern part of the Donbass depression, in bedded salt formations. Argillaceous formations of sufficient thickness are spread over the southwestern slope of the east-European platform (Volyn-Asov plates, Cambrian, Oligocene), and in the Donbass-Dnieper-Donets depression. In the latter locations, detailed geological investigations have not yet been carried out.

In the course of regional studies, several candidate sites have been chosen. In the northern part of the Korosten pluton, two favourable massifs (subzones) have been selected, i.e. Luginy and Volodarsk-Volyn massifs. The Luginy massif is composed primarily of granite-rapakivi, and two sites within its limits have been chosen. Eight salt domes that are potentially favourable for RAW disposal have been selected within the boundaries of the Dnieper-Donets depression.

On the basis of the mining/geology models and considering the criteria mentioned above, a ranking of selected sites has been made. Crystalline formations within the Korosten pluton have been ranked as follows: (1) first priority for the Pribytkov and Doroginby sites in the Luginy massif; and (2) second priority for the Novo-Borovaya and Zankovo sites in the Volodarsk-Volyn massif. Within the limits of the Dnieper-Donets depression, sites have been ranked as follows: (1) first priority for the Kaplintsy, Isachki and Yatsyno-Logoviki salt domes in the northeastern marginal zone; (2) second priority for the Dmitrievka, Siniovka, and Romny salt domes in the southwestern marginal zone; and (3) third priority for the Aleckseevka salt dome in the southeastern part of the depression.

Two zones of Permian bedded salt formation have been studied in northwestern Donbass.

25.4 CONCLUSIONS

As a result of completing the initial stage of the R&D programme, certain regional studies have been carried out. The regional studies resulted in the selection of favourable zones of crystalline formations in the Ukrainian shield and salt formations in the Dnieper Donets depression (as well as in northwestern Donbass, where technogenetic activities have to be considered.) Several candidate sites have been selected in favourable zones. The completion of this initial stage leads to the next stage of specialized exploratory geological/geophysical investigations. This stage is much more complicated and much more expensive.

The initial stage of investigations was completed during 1993-95. Such rapid advances were possible due to a thorough understanding of the geology of the territory of Ukraine, the excellent work of the scientific team, and the availability of results from world experience in the field of RAW in advanced countries (USA, France etc.).

Ukraine possesses scientific and technological capabilities sufficient for the effective completion of the necessary R&D related to exploration and URL construction. But the actual economy in the Ukraine provides no reason for optimism that financing sufficient for an effective realization of such an expensive program will be forthcoming. Thus, the possibilities for program support will depend upon a significant increase in national funds and the organization of international cooperation.

The Institutions involved in R&D programmes in waste isolation (as well as the Ministry of Environment Protection and Nuclear Safety) and the State Committee on Nuclear Energy Utilization, as a sponsoring institution, have initiated an annual international conference, "Isolation of RAW in Geological Formations." The first conference was held September 20-24, 1994 in Kiev. This conference has revealed the interest of eastern European countries (Poland, Slovakia, the Czech Republic, Russia, Hungary, Slovenia, Belarus, etc.) in a program of cooperation. A second conference is scheduled to be held in 1995.

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Chapter 26

The Investigations of the Geology and Hydrogeology at Sellafield in the United Kingdom

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26.1 Introduction

United Kingdom Nirex Limited (Nirex) is responsible for providing and managing a national disposal facility for solid intermediate-level (ILW) and low-level (LLW) radioactive waste. Such wastes have been produced in the UK for over 40 years and have come from the nuclear power industry, medical and defense establishments, as well as from other research and industrial sources. UK Government policy is to dispose of these wastes in a deep underground repository. Similar policies are followed by other countries which produce substantial quantities of long-lived radioactive waste.

Following an extensive site selection exercise, Nirex announced in 1989 that it would investigate, initially, sites at Dounreay in Caithness and Sellafield in Cumbria, to establish their suitability as safe locations for a deep disposal facility for ILW and LLW. Initial boreholes and other geological and geophysical surveys subsequently indicated that the geology at both sites had the potential to meet the demanding safety requirements for a deep repository. In July 1991, Nirex announced that it was to concentrate its further investigations at Sellafield. Given that there appeared to be little otherwise to distinguish between the overall suitability of the two sites, transport of waste and the associated costs were major considerations in this decision; an estimated 60 per cent by volume of the radioactive waste destined for the repository arises from British Nuclear Fuels' operations at Sellafield.

This paper presents a broad overview of the investigations carried out at Sellafield, up to approximately the end of 1994 and of the results obtained. A description is provided of the strategy being adopted for the continued investigation of the site. Descriptions of the geology, hydrogeology and geochemistry studies at Sellafield are provided by Michie (1996), Sutton (1996), and Bath, et al., (1996).

26.2 Scope of Investigations

26.2.1 Scientific Approach

Nirex has adopted a systematic scientific approach to the design and implementation of the investigations. A wide range of technical specialists and techniques have been used in the conduct of the work and care has been taken to avoid undue reliance on any single technique in the interpretation of the ground conditions. The quality of the work being undertaken by Nirex has been recognized by independent reviewers, for example, by the Royal Society Study Group (1994) and RWMAC (1994).

Nirex makes information from the investigations widely available through publications and presentations. A series of papers on the geology of the Sellafield area were presented at a meeting of the Yorkshire Geological Society in late 1993) and subsequently published (Holliday and Rees, 1994). A second meeting on the hydrogeology was held at the Geological Society Apartments in May 1994, the papers presented at this meeting have been submitted for publication. Nirex also releases a significant number of detailed reports on the results of the investigations (For example: Nirex, 1992 Nirex, 1993a-i, Nirex, 1994a-b). An independent panel of university professors carries out review of the work undertaken by Nirex. The first Annual Report of this review panel was released in December 1994.

26.2.2 Areas of Study

The studies carried out in West Cumbria have been contained within three areas (Fig. 26.1):

1. An area onshore and offshore (A) of approximately 60 km by 65 km for which information has been gathered on geological features which might have relevance to a repository safety assessment, using exist-

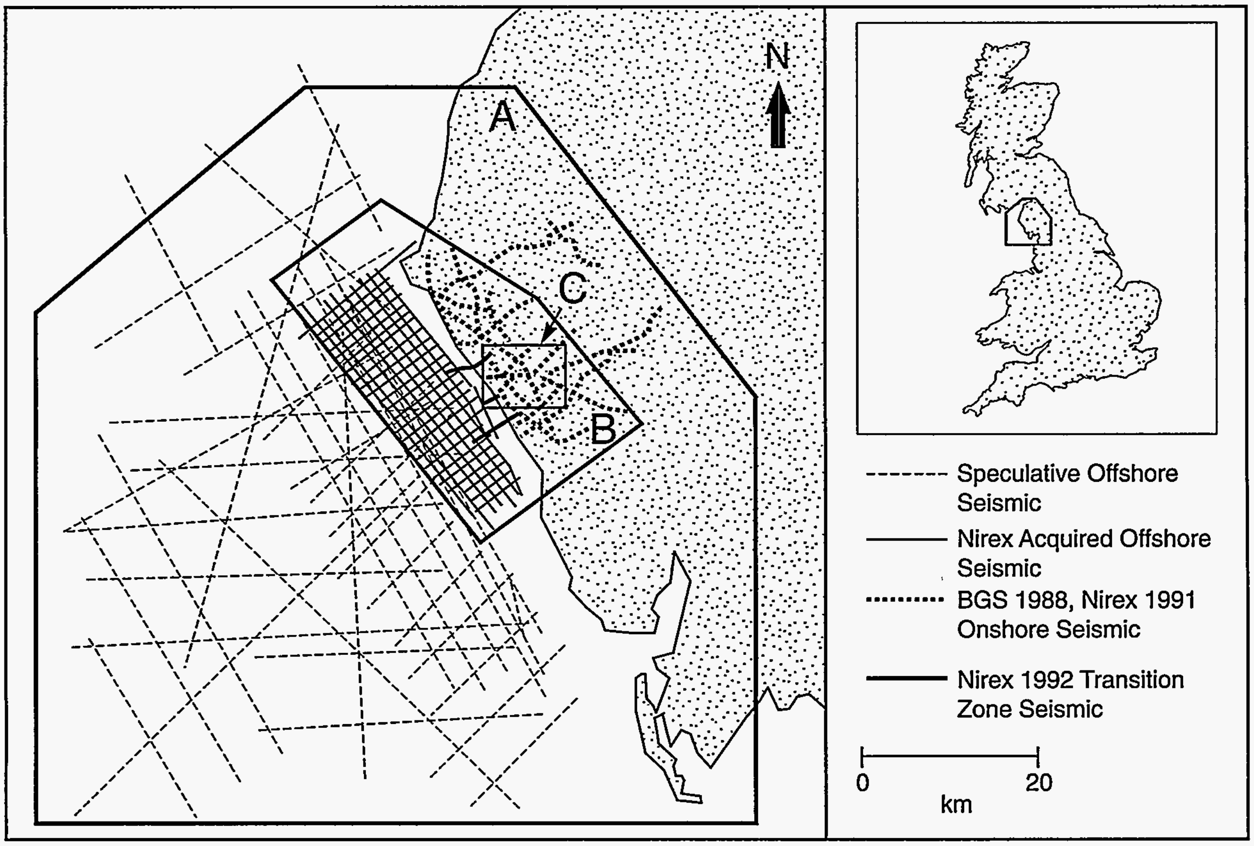

Figure 26.1. Investigation areas showing location of seismic surveys.

ing published sources of information and commercially available offshore seismic survey data. Additional data, including structural geological data relevant to seismic hazard studies, were collected from wider areas.

2. An area (B) of approximately 20 km by 30 km within which geological features have direct relevance to the repository. Within this area Nirex has commissioned new geological, geophysical and hydrogeological investigations. These investigations have been supplemented by study of data from past mining activities.
3. An area (C) immediately around the potential repository covering an area of approximately 50 km^2 and within which all the Nirex deep boreholes are located.

26.2.3 Regional Surveys

The extent of the regional geophysical surveys commissioned or acquired by Nirex is shown in Figure 26.1. These surveys have included some 1,950 line kilometres of seismic reflection, both onshore and offshore, and 8,500 km of airborne magnetic and radiometric surveys. Gravity data has been collected along many of the seismic lines. Geological mapping has been carried out by the British Geological Survey and regional surveys of near-surface hydrogeological features have been commissioned, as have remote seismic studies. Monitoring of springs, river gauging and meteorological observations are continuing. A programme of work to characterize the Quaternary deposits of the area has commenced.

26.2.4 Boreholes

By December 1994 Nirex had drilled twenty one deep boreholes (Fig. 26.2). Many of these were around 1,000 metres deep, with the deepest, Borehole 2, extending to 1,950 metres depth. Several phases of drilling have been completed, namely:

- An initial pattern of boreholes (Boreholes 1, 2, 3, 4, 5, 7, 10, 11, 12 and 14) to obtain an understanding of

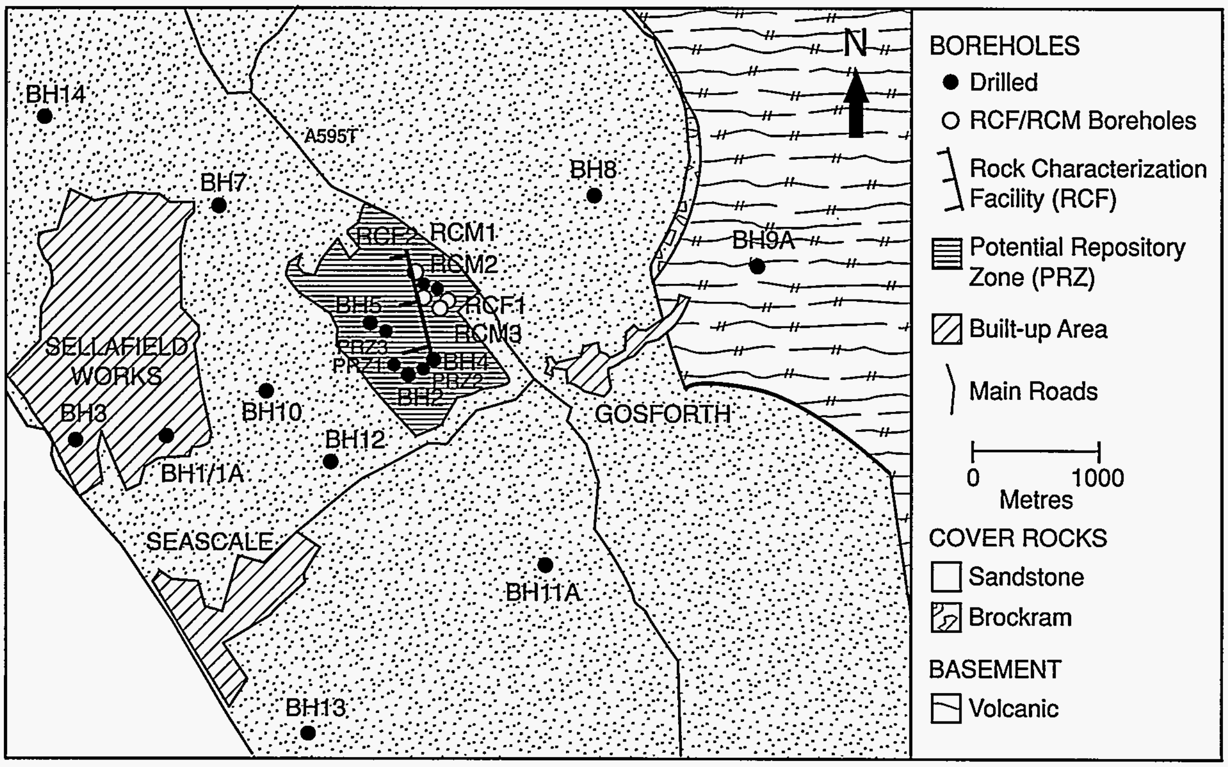

Figure 26.2. Schematic location of Nirex boreholes within the Sellafield site area, December 1994.

the regional geological and hydrogeological setting of the site.

- A subsequent series of boreholes to investigate specific aspects of the site. Boreholes 8 and 9 have been drilled in the upper part of the catchment to investigate groundwater recharge and Borehole 13 has investigated the area to the south where the thickness of the Permo-Triassic sequence increases markedly.
- A series of six boreholes (RCF1 to 3) and RCM1 to 3 have been drilled in the area of the proposed Rock Characterization Facility to characterize the ground in advance of underground excavation and to permit installation of a groundwater monitoring system close to the proposed shafts (Nirex, 1994b).
- Inclined boreholes are being drilled to characterize parts of the potential repository zone. Boreholes PRZ2 and 3 are completed; PRZ1 has still to be completed.

Two boreholes, drilled by others several decades ago for mineral exploration purposes, have been instrumented to supplement the groundwater monitoring system.
The majority of the drilling carried out by Nirex has been to obtain continuous core which is used for detailed characterization of the rock penetrated. Geophysical logging is also carried out to determine rock properties and particularly to provide information on the characteristics of the fractures which occur in the rocks.

Hydrogeological testing is carried out in the boreholes to determine groundwater pressures and the hydraulic conductivity of the rocks, that is, their ability to transmit water. Testing is carried out during breaks in the drilling and after completion of drilling to investigate the hydraulic properties of the rocks at a range of scales (Fig. 26.3).

Sampling of groundwater and analysis of samples is routinely undertaken during drilling and subsequently. Special measures are taken to reduce the levels of contamination of the groundwater by drilling fluids and to quantify the extent of any contamination to permit the determination of groundwater chemistry.

The completed boreholes are also used for undertaking specialist testing programmes. Examples include cross-hole seismic tomography and cross-hole hydraulic testing. A major programme of pump testing to measure the responses of the groundwater system over a wide area to

Field Activity	Scale	Hydraulic Characteristics	Hydraulic Connections to Overlying Units	Calibration/ Validation	Transport Processes
Standard Well Testing	10m	●	●		
Borehole 2/4 crosshole	100-500m	●	Partially		
Fracture Network Testing	10-100m	●			
Short Interval Testing	1-5m	●			
RCF3 Pump Test	Up to km		●	●	
PRZ Monitoring Network	All		●		
Tracer Tests	10-100m			●	●
RCF	All			●	●

Figure 26.3. Types of hydrogeological testing.

pumping from a central borehole is currently in progress.

26.2.5 Long-Term Monitoring

Most of the boreholes have now been converted for long-term monitoring of groundwater pressures by the installation of Westbay multi-level groundwater monitoring systems. These systems permit measurements of groundwater pressures within selected sections of the boreholes. Many sections are now equipped with automatic logging systems which provide measurements of pressures at two minute intervals to a high level of precision (Nirex, 1994a).

Nineteen boreholes have Westbay strings installed, and a further borehole is equipped with an alternative monitoring system. Some of the monitoring strings are amongst the most complex and deepest instrumentation systems of their type ever installed.

The monitoring network is designed to establish baseline groundwater conditions and to provide the means for monitoring the response of the groundwater system to induced perturbations, such as from cross-hole testing, pumping tests and RCF shaft excavation.

26.3 SUMMARY OF RESULTS

26.3.1 Geology

The proposed repository host rock at Sellafield comprises the volcanic rocks of the Ordovician Borrowdale Volcanic Group. Within the potential repository zone, the top surface of volcanic rocks is at a depth of 400 to 600 metres, occurring beneath the immediately overlying Permian breccia, the Brockram (Fig. 26.4). This is in turn, overlain by the Triassic Sandstones of the Sherwood Sandstone Group. The top of the volcanic rocks dips to the west such that at the coast they are some 1,600 metres below the surface. On approaching the margins of the East Irish Sea Basin, the Sherwood Sandstone Group is underlain by a thicker sequence of Permian rocks comprising the St. Bees Shale, the St Bees Evaporite and the Brockram. These are in turn underlain by the Carboniferous Limestone which rests unconformably on the Borrowdale Volcanic Group rocks (Michie, 1996; Holliday and Rees, 1994).

The rocks have been subjected to numerous periods of faulting and folding during their geological history. The distribution of the various formations at depth and the locations of the faults which cut them have been defined primarily by interpretation of the seismic reflection data, calibrated by the deep boreholes and utilizing mine plan data for the area to the north of Sellafield. Structure contour maps covering areas A and B have been generated for all the major geological boundaries within the sequence (Nirex, 1993a, b). Within the potential repository zone, additional detail is now being added based upon further boreholes, seismic reflection surveys, cross-hole tomography surveys between sets of co-planar boreholes and complex analysis of existing vertical seismic profiling (VSP) data.

26.3.2 Hydrogeology

Much of the work being undertaken by Nirex at

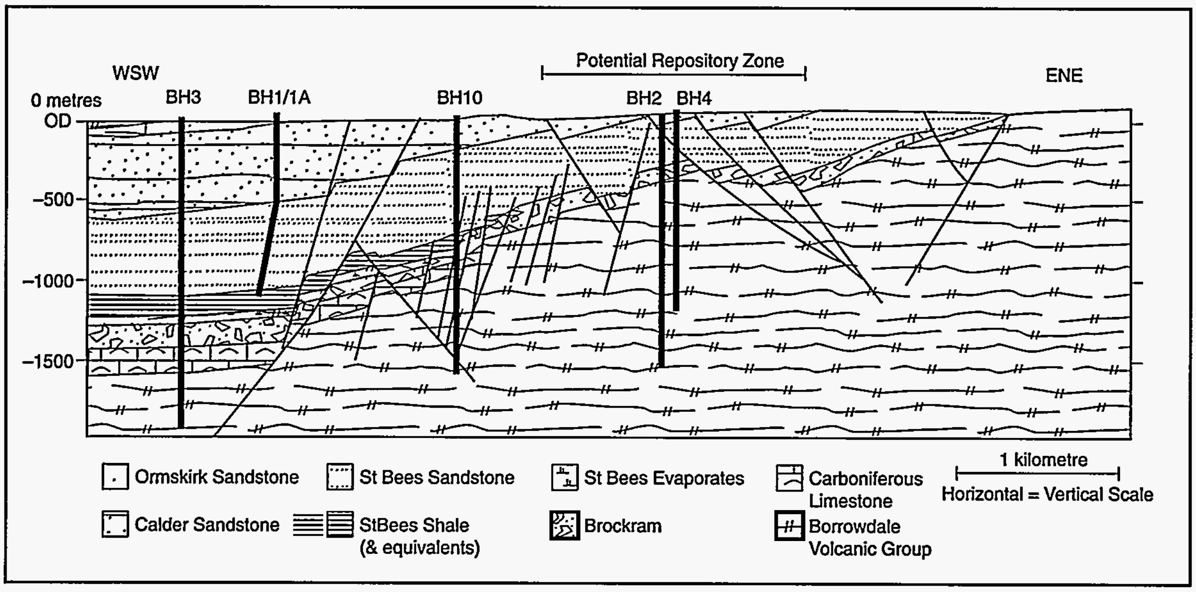

Figure 26.4. Schematic geological WSW-ENE cross-section through the Sellafield area.

Sellafield has been focused on determining the hydrogeology of the site, given that the flow of groundwater is recognized as the dominant mechanism for transport of radionuclides from a repository back to the surface (Nirex, 1993i; Black and Brightman, 1996).

Measurements have been made of hydraulic properties, especially heads and conductivities in boreholes. Having recognized that the flow of groundwater through the volcanic rocks is principally through fractures, effort has been directed towards characterizing those fractures which are both open and inter-connected such that they are hydrogeologically significant.

Geochemical studies of the groundwater have provided an independent record of past flow and mixing, and hence geochemical studies have featured prominently in the work undertaken (Bath, et al., 1996). Finally, numerical modeling has been extensively used to develop the understanding of the processes which are controlling-groundwater flow (Heathcote, et al., 1996).

The hydraulic conductivity of the rocks was initially determined in the boreholes using 50 metre long contiguous sections. Within the Borrowdale Volcanic Group the conductivity values are typically very low (Fig 26.5) with half the values measured over 50 metre lengths in the boreholes being less than $1x10^{-10}$ ms, including tests over faulted and fractured zones.

In order to identify the distribution of the hydrogeologically significant fractures in the parts of the sequence dominated by fracture flow, production tests have been carried out over the full lengths of boreholes, often in a series of stages. Inflow of water into the borehole is induced by imposing a drawdown in the order of 100 metres head and identifying flow zones by production logging. In many cases flows are so low as to preclude the effective use of spinner logging, zones only being identified from differential temperature and conductivity logging (Fig 26.6). Individual fractures, or groups of fractures, which carry flow are characterized by reference to the core logs and the borehole imaging geophysical logs.

Most of the fractures intersected by the boreholes have no detectable flow. Flowing fractures are therefore relatively widely spaced. Just over 150 have been identified in over 20,000 metres of drilling. Studies are currently being undertaken to characterise them.

Although fractures encountered in particular boreholes can make a major contribution to the conductivity of the rock mass in the immediate vicinity, it is the extent to which conducting fractures are connected which will determine groundwater flow in the Borrowdale Volcanic Group. Cross-hole seismic tomography has helped to define the geological structure between adjacent boreholes.

The extent of the connectivity is being examined using single borehole fracture network testing, cross-hole

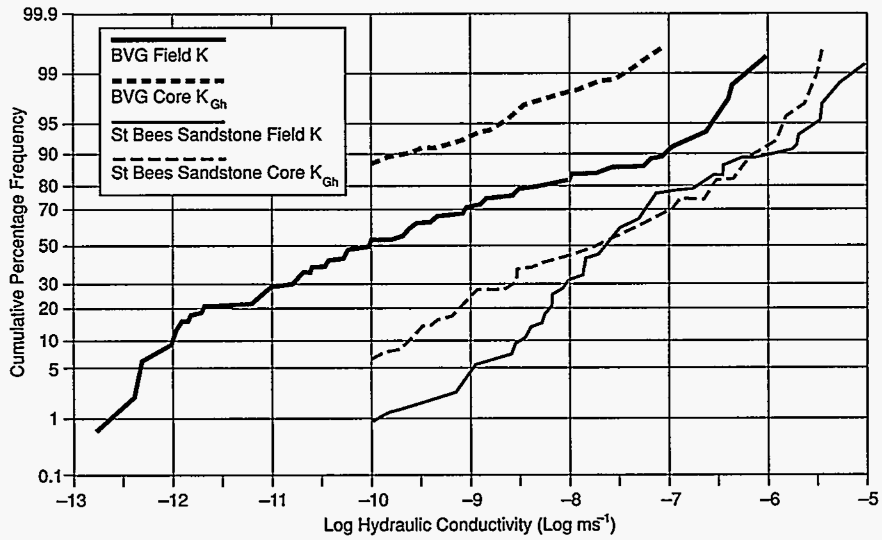

Figure 26.5. Summary of hydraulic conductivity values.

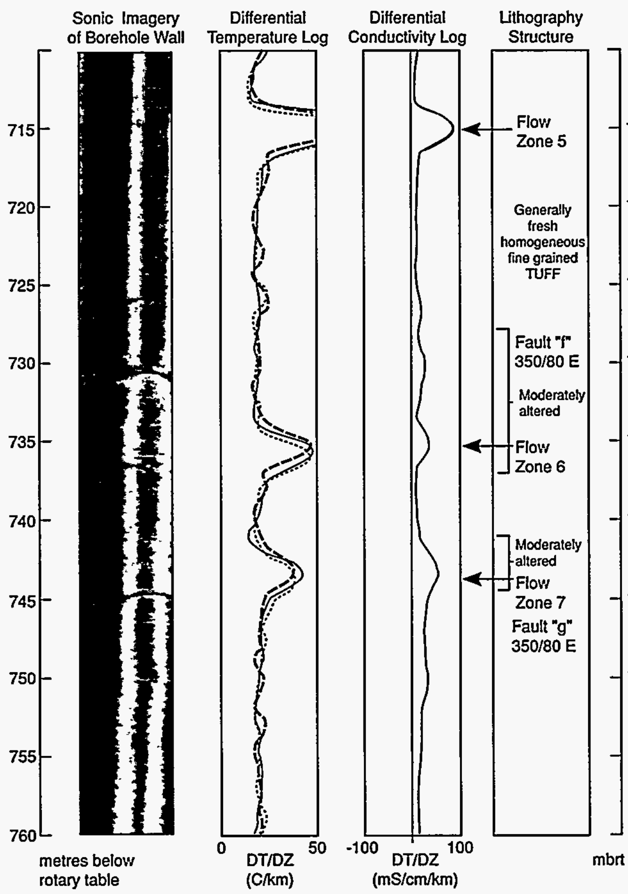

Figure 26.6. Identification of flow zones in Borehole 2.

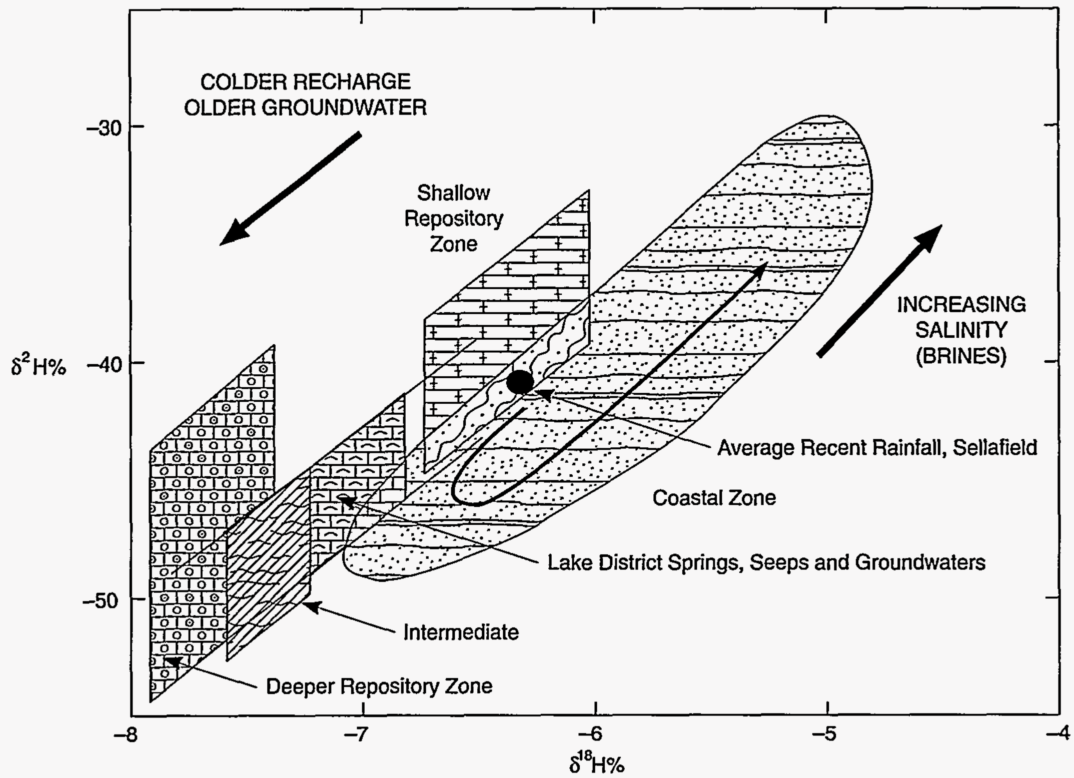

Figure 26.7. Stable isotopic discrimination between groundwaters with different origins.

hydraulic testing and by a series of large scale pump tests. Preliminary results from recent testing suggest that the fracture network may be less well connected than previously considered to be the case.

26.3.3 Geochemistry

Geochemical studies (Bath, et al., 1996) are carried out for three reasons: (a) to support the development of a conceptual model of the present-day hydrogeology, (b) to investigate how the groundwater system has evolved over time; and (c) to characterize the baseline hydrochemical conditions to support other studies.

Considerable progress has been made and the present dataset has contributed substantially to the construction of a conceptual model of the hydrogeological system, on which numerical modeling can be based. Some hydrochemical aspects of the conceptual model (particularly salinity sources and mixing zones) will provide specific tests of the adequacy of numerical modeling.

The palaeohydrogeology of the area is dominated by its location on the margin of the East Irish Sea Basin. The influence of basinal brines has been a feature of the deep sediments and the Borrowdale Volcanic Group basement in the west of the area for considerable geological time. Within the potential repository zone, a range of analyses including stable isotopic and noble gas temperature data for groundwaters in the Borrowdale Volcanic Group basement suggests that the waters at depth are clearly distinguishable from the shallow groundwaters and from modern rainfall (Fig. 26.7), and that the deeper waters probably have long residence times. This is a consistent pattern shown by several independent determinants and studies (Bath, et al., 1996).

26.3.4 Hydrogeological System

The current conceptual model of the hydrogeological system is illustrated in diagrammatic form in Figure 26.8. The three component parts of the system: the Irish Sea Basin Regime, the Hills and Basement Regime, and the Coastal Plain Regime are essentially as defined in mid 1992 (Nirex, 1992), although greater confidence in this model has been obtained with the availability of geochemical data.

The conceptual model and its evolution are supported, not only by geological, hydrogeological and geochemi-

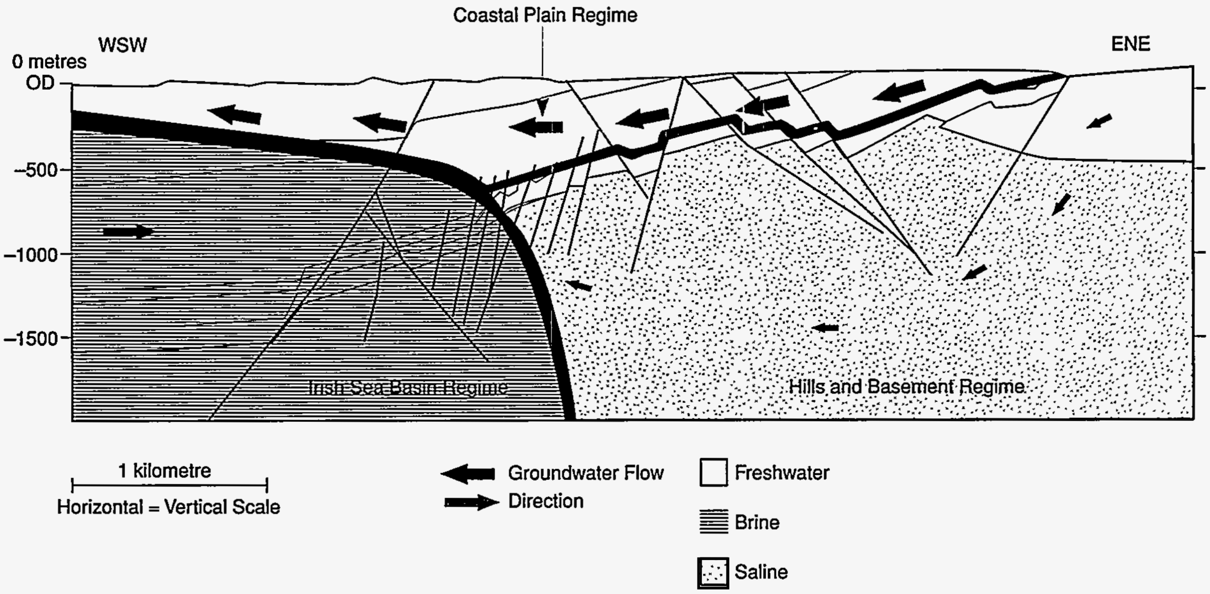

Figure 26.8. Current conceptual model of the groundwater system in the Sellafield area.

cal studies, but also by a range of numerical modeling studies which have served to examine and test a series of concepts concerning the behavior of individual components of the total system (Heathcote, et al., 1996).

26.4 FURTHER STUDIES

The investigations carried out to date from the surface are providing Nirex with a good understanding of the geological and hydrogeological conditions at Sellafield as they affect the decision on whether or not the site will be suitable for the construction of a repository to meet the stringent regulatory safety targets. Further investigations from the surface are in progress or are planned to further characterize specific features of the site. In particular, the pump test in Borehole RCF3 and the continued monitoring of groundwater conditions utilizing the installed instrumentation system are important components of this forward programme.

Investigations from the surface are, however, unable to resolve all the remaining uncertainties regarding the characteristics of the site. Nirex, in line with similar agencies in other countries, considers that a phase of investigations carried out underground from a suitably constructed experimental facility is a logical and essential extension to characterization from the surface. For this reason, Nirex has applied for planning permission to construct an underground Rock Characterization Facility (RCF) at the Longlands Farm site at Sellafield. This application is the subject of an appeal by Nirex against refusal, by Cumbria County Council, of planning permission.

26.5 CONCLUSIONS

The following conclusions are drawn from the investigations carried out at Sellafield:

1. An extensive programme of investigations has been carried out by Nirex. Various independent reviewers have commented on the high quality of the work undertaken.
2. The geological succession and structural geology of the site has been determined in significant detail.
3. Cross-hole seismic tomography has demonstrated that the geological structures can be mapped between boreholes. This observation is providing added confidence regarding the definition of the geological structure within the Borrowdale Volcanic Group and its influence on the hydraulic conductivity of the rock mass.
4. Preliminary quantitative assessments have been made of the distribution of hydraulic conductivity values in all the major hydrogeological units. Values measured in the Borrowdale Volcanic Group are typically low.
5. A limited number of individual fractures has been identified in the Borrowdale Volcanic Group along which groundwater flows. These fractures form a network which controls the flow of water through the rocks. Nirex is currently assuming that the fracture

network is well connected. Preliminary results from recent testing suggest that the fracture network may be less well connected than previously considered to be the case.

6. Geochemical and isotopic analysis of groundwater samples have assisted the development of the hydrogeological conceptual model and are helping to give some indication of the age and provenance of the groundwater within the Borrowdale Volcanic Group rocks. Evidence is suggesting that the deeper groundwater in the potential repository zone is old.
7. Good progress has been made with the investigations, and with the interpretation and modeling studies which follow on, to determine whether or not the site is suitable as a potential repository. However, much work remains to be done to resolve uncertainties and to develop confidence in the understanding of the site and the models which are constructed to represent its behavior. Construction of an underground Rock Characterization Facility forms a logical and essential continuation to the investigations carried out from the surface to progressively reduce uncertainty and to provide the information necessary to determine the suitability of the site for construction of a deep radioactive waste repository to meet the stringent regulatory safety requirements.

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CHAPTER 27

HIGH-LEVEL RADIOACTIVE WASTE MANAGEMENT IN THE UNITED STATES BACKGROUND AND STATUS: 1996

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27.1 INTRODUCTION

The United States high-level radioactive waste disposal program is investigating a site at Yucca Mountain, Nevada, to determine whether or not it is a suitable location for the development of a deep mined geologic repository. At this time, the United States program is investigating a single site, although in the past, the program involved successive screening and comparison of alternate locations. The United States civilian reactor programs do not reprocess spent fuel; the high-level waste repository will be designed for the emplacement of spent fuel and a limited amount of vitrified high-level wastes from previous reprocessing in the United States. The legislation enabling the United States program[1] also contains provisions for a Monitored Retrievable Storage facility, which could provide temporary storage of spent fuel accepted for disposal, and improve the flexibility of the repository development schedule.

Yucca Mountain is a mountainous ridge located in the southwestern United States (Fig. 27.1) in the southern Great Basin, the largest subprovince of the Basin and Range physiographic province of the United States. The Basin and Range province is that area of southwestern North America that is characterized by more or less regularly spaced subparallel mountain ranges and intervening alluvial basins formed by extensional faulting. The regional climate of the southern Great Basin is typically hot and semi-arid. Generally, the geology of the province can be described as a late Precambrian and Paleozoic continental margin assemblage that has been complexly deformed by late Paleozoic and Mesozoic orogenies. Western portions of the province are broadly overlain by Cenozoic volcanic rocks; the distinctive physiography is largely a product of the most recent phase of extensional deformation. The alluvial basins are characterized by low rainfall, high evapotranspiration, ephemeral streams and closed hydrologic systems, evidenced by the absence of drainage external to the basins[2]. Characteristics such as these were important waste isolation considerations in the selection of Yucca Mountain for site characterization.

The repository design concept is a mined excavation at a depth of approximately 300 meters below the crest of Yucca Mountain and at a distance of approximately 300 meters above the regional groundwater table. The site is in silicic volcanic rocks, comprising alternating layers of welded and non-welded volcanic tuffs. The non-welded tuffs underlying the proposed repository horizon contain layers that are extensively zeolitized. The strategy for waste isolation relies on both engineered and natural barriers to provide defense in depth. The strategy for long term waste isolation places primary reliance on, and takes advantage of, the natural barriers, which include the aridity of the site, the unsaturated character of the host rock, and the deep regional water table. All indications are that these geologic conditions have been both spatially and temporally stable for many millions of years.

The United States high-level waste disposal program is managed separately from activities related to the management of transuranic waste from national security activities or commercially-generated low-level wastes. The transuranic waste program in the United States is also pursuing development of a mined geologic repository for disposal[3]. The Waste Isolation Pilot Plant, located near Carlsbad, New Mexico, is constructed in a salt formation.

27.2 LEGISLATIVE BACKGROUND

The high-level waste disposal program in the United States evolved through several different approaches between 1955 and 1982. In 1955, the National Academy of Sciences was asked to recommend a strate-

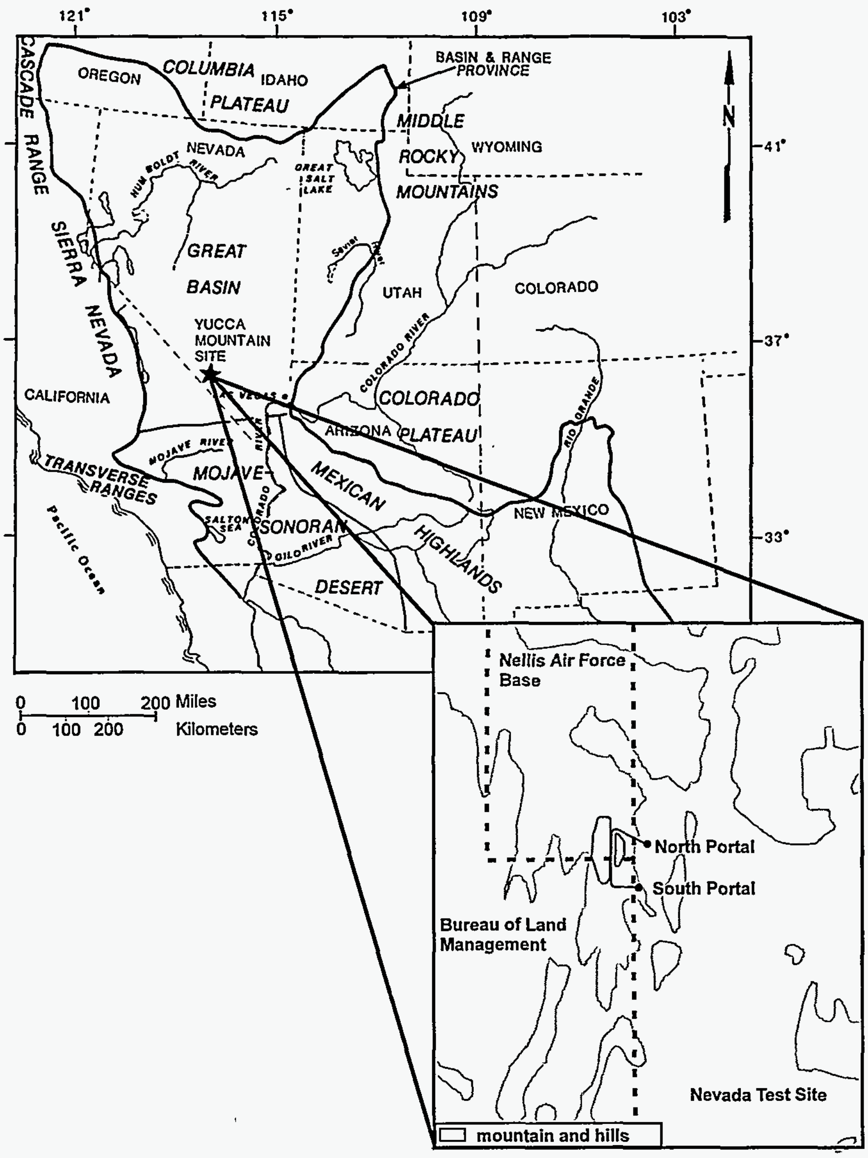


Figure 27.1. Location map for Great Basin physiographic province and Yucca Mountain Site.

gy for the disposal of liquid chemically hazardous radioactive wastes resulting from the reprocessing of spent nuclear fuel. The National Academy of Sciences recommended disposal in salt formations, to take advantage of the evident geologic stability and isolation from water. Subsequently, the United States program recognized the advantages of other rock characteristics, and began a program of national screening in the late 1970s. In 1982, the United States Congress passed the Nuclear Waste Policy Act (the Act) [4], setting forth an integrated plan for the disposal of commercially-generated spent fuel and high-level wastes from military reprocessing . The Act established responsibilities, schedules, and a funding mechanism whereby the users of nuclear power would pay for the disposal of the generated wastes. A portion of the wastes generated through various United States defense programs are included in the provisions of this law.

Under the Act, the United States Department of Energy was assigned responsibility for management of the civilian high-level waste program, the United States Nuclear Regulatory Commission was assigned responsibility for approving or disapproving a license to construct a repository and amendments to construct, operate, and close it, and the United States Environmental Protection Agency was assigned responsibility for promulgation of generally applicable standards for protection of the general environment from offsite releases from radioactive material in repositories. The United States Environmental Protection Agency promulgated its standard[5] in 1985. This standard is not explicitly health based; the postclosure behavior of a repository licensed under this regulation would be judged against total cumulative releases at the end of a 10,000 year period. In 1987, a court challenge, based to some degree on inconsistency with other United States Environmental Protection Agency regulations, led to a remand of the standard. The United States Nuclear Regulatory Commission had already promulgated its procedural and technical requirements[6] by 1983. Those requirements, which implement rather than duplicate the United States Environmental Protection Agency release standards, address the concepts of multiple barriers and defense in depth, placing specific requirements for postclosure performance of the repository on components of both the engineered and natural subsystems.

Following completion of site characterization, the United States Department of Energy will submit initial documentation for a licensing hearing to authorize construction of the repository. Authorization of construction will be based in part on an understanding of the long term performance of the proposed repository. The United States Nuclear Regulatory Commission requirements embody a phased approach to construction and emplacement of high-level wastes in a repository. After sufficient construction to affirm that site conditions and underground excavation response are within the limits specified in the license to construct, the applicant submits documentation for a hearing for a license to receive and possess wastes. Amendment of the license to allow the repository to receive and possess wastes marks the first point in time that high-level wastes can be emplaced for disposal in the repository. With the approval of the United States Nuclear Regulatory Commission, up to ten metric tons of spent fuel could be emplaced for testing purposes during site characterization. Other than this limited amount of waste allowed for use by the Nuclear Waste Policy Act for purposes of site characterization (the provision was not invoked by the United States Department of Energy for Yucca Mountain), radioactive wastes are not permitted to be emplaced until this license is received. After operation of the repository and a defined period of monitoring, an application is submitted for a license amendment to decommission and then permanently close the repository.

The intention of the Program created by the 1982 Nuclear Waste Policy Act[4] was to characterize multiple sites and recommend sites for development as repositories. The Act envisioned the need for two repositories in the commercial waste program; accordingly, it directed the United States Department of Energy to undertake two repository characterization programs. The Act specified that the United States Department of Energy develop guidelines and prepare Environmental Assessments to be used as the basis for selecting each set of three sites to be characterized. The Act also explicitly required the preparation, and submittal for review by the United States Nuclear Regulatory Commission, of a Site Characterization Plan. It was the intent of the Act that following completion of the characterization of three candidate sites for each repository, Environmental Impact Statements would be prepared and serve as the basis for the recommendations of the sites for which the United States Department of Energy would apply for a license to construct a repository. In 1987, the United States Congress amended the Act. The Nuclear Waste Policy Amendments Act[1] selected Yucca Mountain, Nevada, as the single site to be characterized. The amendment also directed the United States Department of Energy to cease work on the second repository program.

The United States Congress recently passed a comprehensive Energy Policy Act[7] that contained provisions that probably will affect the regulations governing a repository at Yucca Mountain. That legislation required the United States Environmental Protection Agency, based upon, and consistent with the findings and recommendation of a study to be undertaken by the United States National Academy of Sciences, to promulgate public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site. These standards are to prescribe the maximum annual effective dose equivalent to individual members of the public. The issue of a dose-based standard for the United States high-level waste program dates back to the United States Environmental Protection Agency's own Science Advisory Board and the National Academy of Sciences[8] noting that such a standard would be appropriate for the United States Program.

The National Academy of Sciences study[9] provided recommendations as to whether a health based standard is reasonable, whether it is reasonable to assume that a system of postclosure oversight, based on active controls, will prevent a risk of breaching the repository, and whether it is possible to make scientifically sound predictions of the probability of human intrusion over 10,000 years. The United States Environmental

Protection Agency currently is in the process of developing a new compliance standard for the Yucca Mountain site that addresses these recommendations.

27.3 YUCCA MOUNTAIN SITE WASTE ISOLATION STRATEGY

The strategy for waste isolation for the Yucca Mountain site consists of reliance on a number of barriers, both natural and engineered, that either are attributes of the site or are engineered in a manner to complement the site attributes[2]. As water is the medium that can dissolve and transport solid wastes, the strategy takes advantage of the paucity of water at the site.

Yucca Mountain is a remote mountainous ridge located in the arid southwestern United States (Fig. 27.2), where rainfall averages approximately 15 centimeters per year. The water table in the vicinity of the Yucca Mountain site is deep, approximately 700 meters below the crest. Placing a repository at a depth of approximately 200 to 300 meters below the surface would leave a distance of several hundred meters between the repository and the water table. The repository would thus be in unsaturated rocks, with water held in place by capillary forces. The stratigraphy at Yucca Mountain consists of alternating layers of welded and non-welded volcanic tuffs. The welded tuff matrix is relatively impermeable; however, the rocks are fractured and will transmit water provided there is a sufficient source. The non-welded tuffs are porous and permeable; however, they tend to form capillary barriers at contacts with the welded tuff units and transmit significant quantities of water only when fully saturated. The repository would be located in a thick welded layer, overlain by a non-welded layer with a welded caprock (Fig. 27.3). Conceptually, this combination should be effective in limiting the amount of water that could eventually reach the emplaced wastes. The rock beneath the repository area includes layers that are conspicuously zeolitized, providing the potential for sorption to be effective in retarding the transport of some radionuclides.

It is intended that the engineered components of the repository complement the natural attributes of the site. The subsurface layout of the repository, as shown in Figure 27.4 and Figure 27.5, would comprise two inclined access ramps, two vertical ventilation shafts, and essentially flat-lying main and waste emplacement drifts.

The waste container is expected to function as the prin-

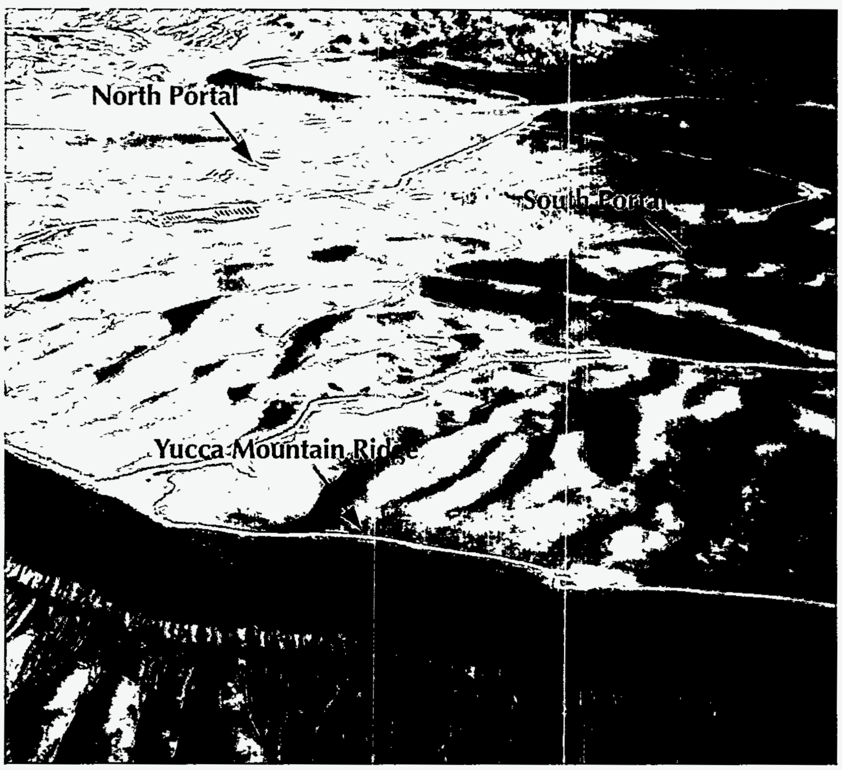


Figure 27.2. Aerial view of Yucca Mountain looking to the northeast.

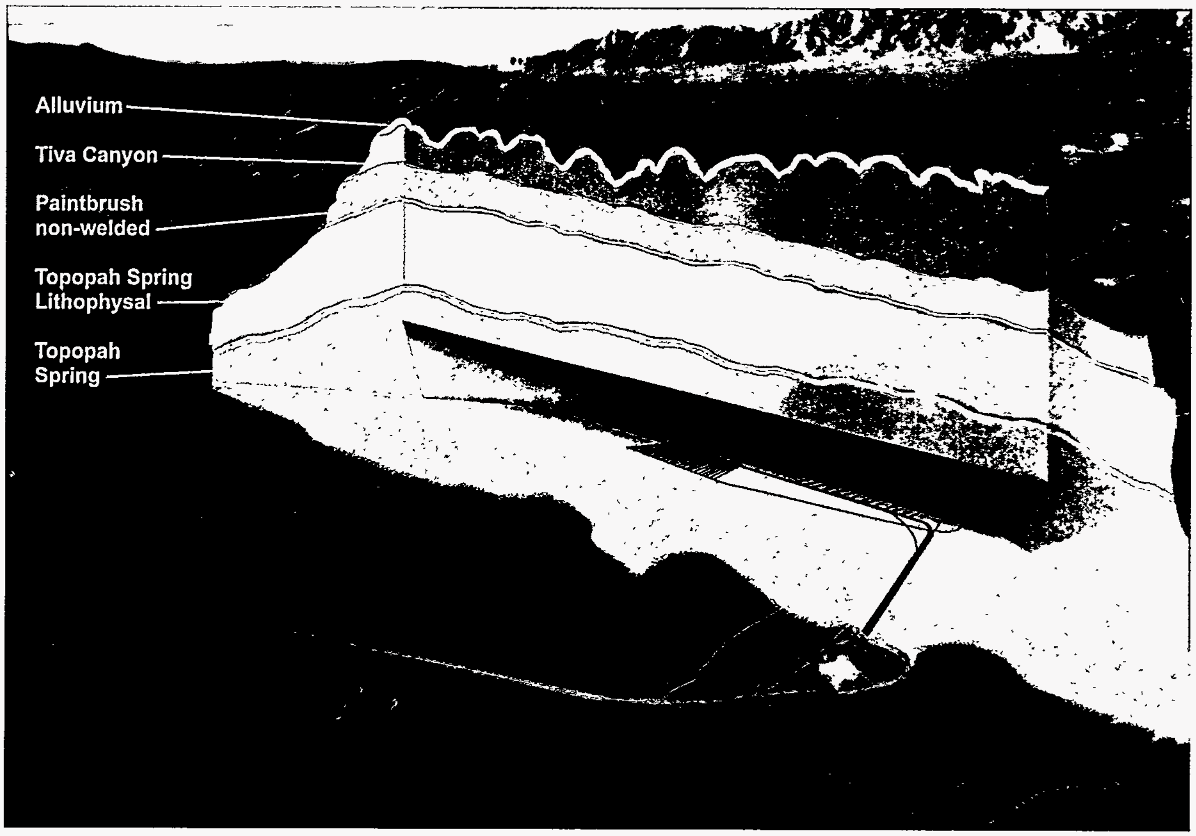


Figure 27.3. Artist's conception of a repository layout at Yucca Mountain illustrating stratigraphy of the mountain.

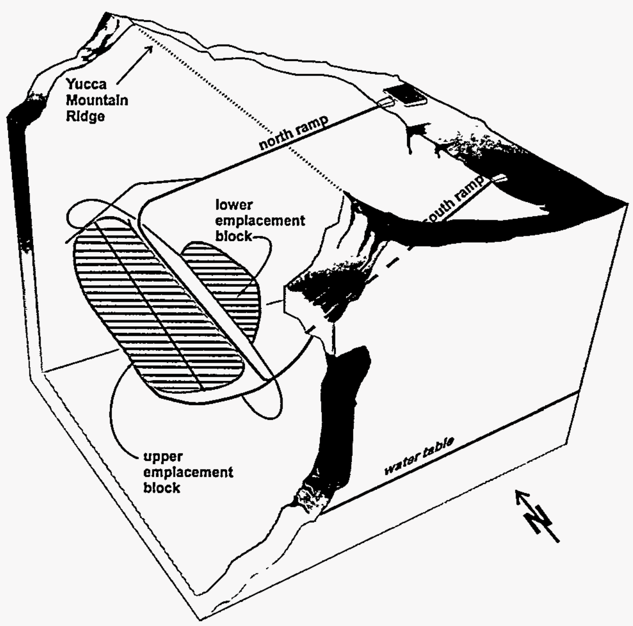


Figure 27.4. The subsurface of the repository will consist of about 250 kilometers (150 miles) of drifts, most of which will be used for emplacing the waste packages. The emplacement drifts will be divided into two areas. The upper block will be large and will lie to the west and slightly above the lower block.

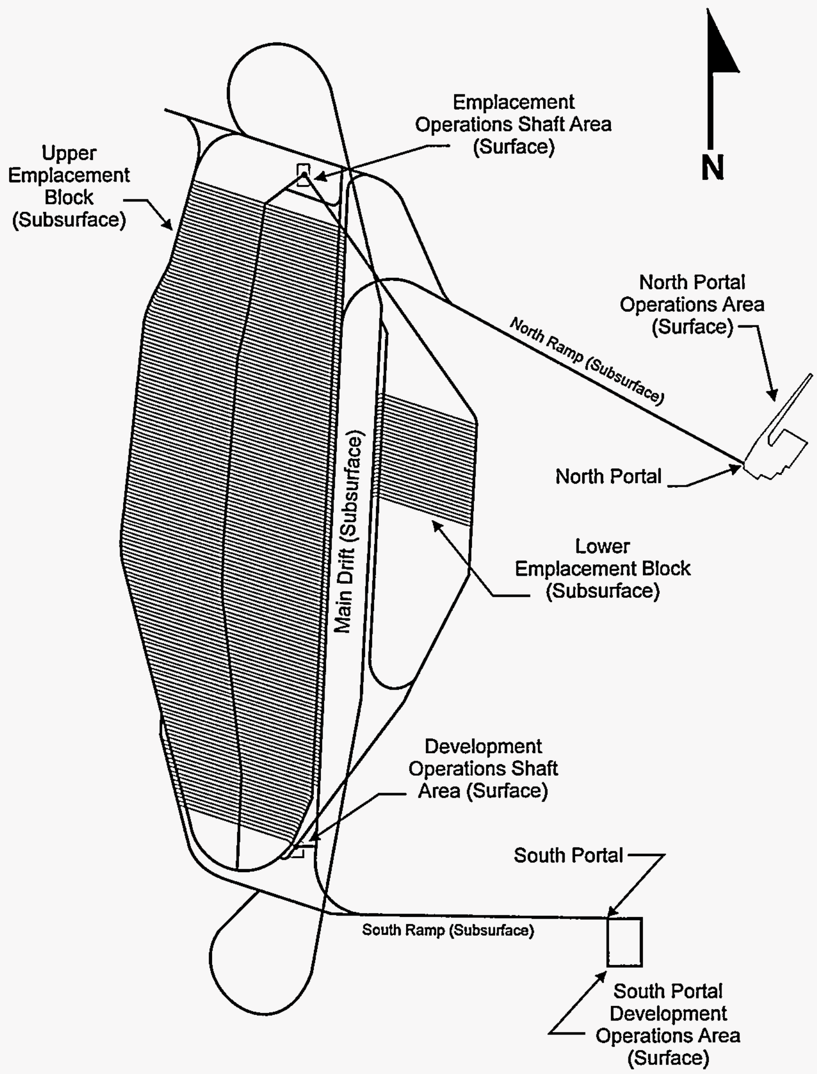


Figure 27.5. Overall repository site map.

cipal barrier to the release of radionuclides from the engineered barrier system. The barrier will be designed to provide substantially complete containment of the wastes for at least the time that heat and radiation emitted by the wastes are at their peak. This occurs early in the postclosure time frame, encompassing the period of approximately 300 to 1000 years following closure. The limited amount of water flowing through the unsaturated zone is expected to enhance the ability of the container to limit the release of radionuclides. Additionally, container materials will be chosen to be compatible with the geochemical properties of the water to prolong container life should water contact it. The waste form itself is also expected to be a barrier to the release of radionuclides. The low probability of early container failure and the small amount of water available are expected to limit the dissolution and leaching of radionuclides from the solid waste materials for at least several thousand years[10].

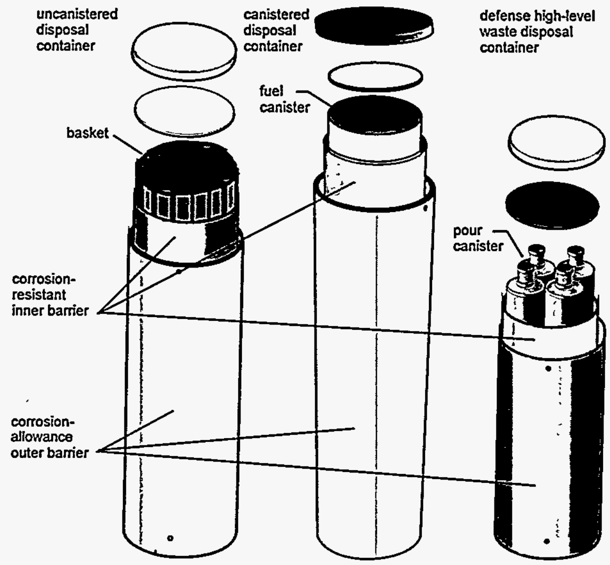
The components of the waste packages, in this relatively dry environment, are intended to confine the waste for thousands of years. The current container designs[11] are deliberately robust; the dual wall design uses a corrosion allowance material outer layer and a corrosion resistant material inner layer to form the walls of the waste packages. The heat from the waste packages is expected to keep the rock, immediately around the emplacement drift, relatively dry for hundreds of years, which should reduce the corrosion rate of the waste packages. The air gap between the container and the host rock is also expected to contribute to limiting the release of radionuclides. Because percolation rates are expected to be low, and because most water is expected to be tightly confined in the rock matrix, little water should cross this air gap. Limited water movement in fractures is possible; however, the amount of water that could potentially contact the containers is expected to be a small fraction of that contained in the rock. As the waste packages and emplacement drifts eventually cool, water could begin to seep back toward the waste packages, especially along fractures.

After the waste packages eventually corrode and deteriorate, and the engineered barrier function is degraded, the natural barriers will provide the primary means of isolation. The various rock layers at Yucca Mountain, due to low water content and movement, are expected to retard the movement of released radioactive material to the accessible environment. Finally, any radioactive material that eventually reached the water table beneath the repository would be diluted, further reducing the potential amounts that could reach the environment.

The current repository conceptual design assumes a relatively high emplacement density of 83 metric tons of initial heavy metal (MTU) per acre. The resulting emplacement scheme divides the subsurface facilities into two sets of waste emplacement drifts called blocks[11]. In this concept, the emplacement drifts would be spaced at 22.5 meters and emplacement drifts would range from about 250 meters to about 600 meters in length. The upper block would cover about 324 hectares, and could accommodate about 11,000 waste packages. The lower block would cover about 69 hectares, and could accommodate up to 2,400 additional waste packages. A total of about 12,000 waste packages are expected to be emplaced[11, 12]. Approximately 90 percent of the radioactive waste will be spent fuel in waste packages, each containing up to 9 metric tons of spent fuel. However, it is currently expected that more than one-fourth of the total number of waste packages will be high level waste, with each package containing approximately 2 metric tons of high level wastes.

The physical arrangement in which waste packages would be placed underground will have an impact on the environmental conditions in the emplacement drifts. After emplacement, heat will raise the temperature of the emplacement drift rock walls. The spacing between the emplacement drifts and the spacing between waste packages within the emplacement drifts will determine how hot the emplacement drift environment and surrounding rock will become. Conceptual designs have been developed for fuel assemblies that have been shipped to the repository in containers (canistered and defense high-level waste disposal containers) that can be placed into an overpack for subsequent emplacement for disposal in the repository. Conceptual designs also have been developed for fuel assemblies that arrive in a shipping container (uncanistered) and must be repackaged for subsequent emplacement for disposal in the repository (Fig. 27.6).

The waste packages would be mounted on rail cars that



Conceptual design characteristics for disposal containers.

	Uncanistered		Canistered		DHLW
	large	small	large	small	
Capacity (number of fuel assemblies or pour canisters)					
PWR	21	12	21	12	—
BWR	44	24	40	24	—
Canisters	—	—	—	—	4
Dimensions in millimeters					
Diameter	1629 (64")	1298/1265* (51" / 50")	1802 (71")	1531 (60")	1709 (67")
Length	5335 (210")	5335 (210")	5682 (224")	5647 (222")	3680 (145")
Weight in kilograms (approximate)					
Empty	31,000 (34 tons)	22,000 (24 tons)	31,000 (34 tons)	25,000 (28 tons)	13,000 (14 tons)
Loaded	47,000 (52 tons)	31,000 (34 tons)	65,000 (72 tons)	47,000 (52 tons)	22,000 (24 tons)

PWR - Pressurized Water Reactor BWR - Boiling Water Reactor
* larger diameter for PWR, smaller diameter for BWR

Figure 27.6. Conceptual designs address disposal containers for fuel assemblies not in canisters, fuel assemblies in canisters, and pour canisters.

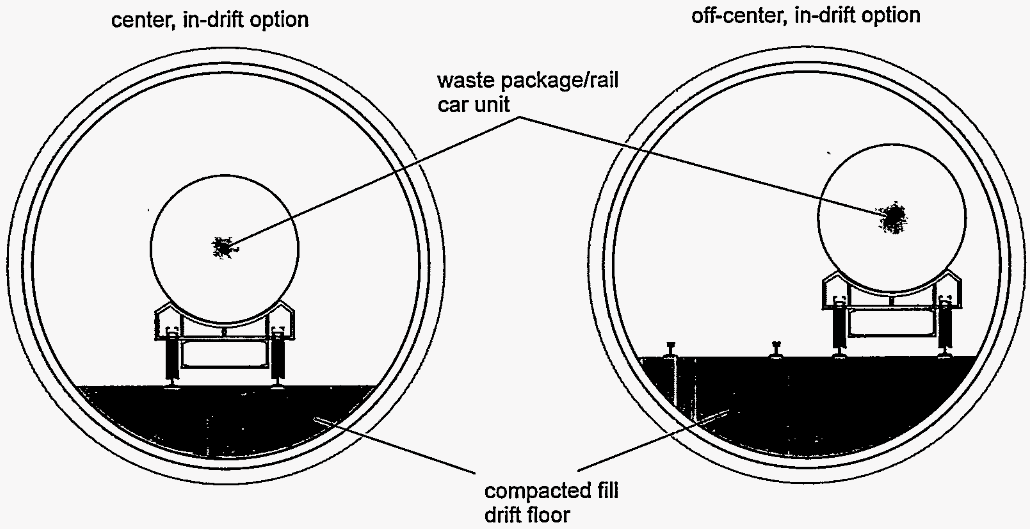


Figure 27.7. Conceptual design options for emplacement drifts: center in-drift, with a single set of rails for placing the waste package in the center of the drift; and off-center in-drift, with two sets of rails for placing the waste package to one side of the drift.

would be placed on rails within the emplacement drift. Once waste packages are placed in an emplacement drift, no human entry into that emplacement drift would be allowed under normal conditions. Two options, illustrated in Figure 27.7, are being studied for emplacement of waste packages within the emplacement drifts. One option, center in-drift, involves a single set of rails running down the center of the drift on which the waste package rail car unit would be placed. The diameter of this emplacement drift would be 5 meters. The other option, off center in-drift, would contain two parallel sets of rails. The rail car would be placed on one set of rails to the side of the drift, and the set on the other side of the drift would provide remote access along the drift for inspecting, monitoring and maintaining the drift, emplaced materials, and equipment. The diameter of this emplacement drift would be 5.5 meters.

Each combination of drift and package spacing, and waste package heat output, will result in a different overall thermal load. Corresponding temperatures will cause various changes in the repository rock and its water content, which can affect waste package performance. Potential changes in rock strength, water content and mineral composition depend primarily on the thermal loading. These changes are being investigated by both field and laboratory testing and by the use of computer modeling techniques. A repository design goal to not exceed wall rock temperatures of 200° C was established[11]. The current conceptual repository design uses a relatively high thermal loading that results in maximum emplacement drift wall temperatures of approximately 155° C at about 40 to 60 years after emplacement, if no ventilation is supplied to the drift after emplacement is completed[12].

As would be expected, this strategy is sensitive to disruptive processes and events, especially those that could modify the dry character of the site, which is central to the compliance arguments. Of particular importance to the strategy are climate changes and processes or events that could enhance infiltration. The effects of repository heating on the conditions of the rock mass are also of concern. A high thermal load may drive moisture away from the canisters for significant periods of time. Conversely, the temperature changes could lead to potentially irreversible changes in the hydrologic and geochemical properties of the zeolites in the rock mass.

Another issue of engineered barrier performance is related to the presence in the waste of gaseous radionuclides such as $^{14}CO_2$. The natural barriers at Yucca Mountain are not likely to be effective in containing gases that could be released through eventual failure of waste containers. Early waste package failure could lead to the release of quantities of $^{14}CO_2$ that could possibly violate the remanded standard for total curies released, even though there would be no significant associated health effects. Although the standard has been remanded, it is still used as a standard for comparison by the program until a new one is in place The waste packages likely could be designed to contain the gases, but the lack of true health and safety protection makes it difficult to justify the costs.

27.4 RESULTS OF SITE CHARACTERIZATION COMPLETED TO DATE

The test program described in the Site Characterization Plan[2] included both surface based and underground test programs, the latter in a facility developed at the base of an exploratory shaft. Review comments on that plan from the Nuclear Regulatory Commission[13] and the United States Nuclear Waste Technical Review Board[14] were addressed[15] and, in so doing, the configuration was changed from shaft access to ramp access. The resulting facility design incorporated requirements to support a test program to address data needed to assess the role of both the engineered barriers and the natural barriers. Heater tests, ranging in scale up to room size, will be conducted over the next several years to investigate coupled thermal-mechanical-hydrological-geochemical processes, moisture movement, mechanical effects, and near field effects. Other underground tests underway or planned include construction tests, which examine excavation methods, measurements of the response of ground support systems, possible emplacement equipment tests, seal component tests, seismic tomography, and rock strength tests. Hydrology and transport will be studied through suites of tests such as large scale permeability tests, radial borehole permeability (gas and water) tests, hydrochemistry tests, mapping, diffusion tests, radionuclide transport tests (using non-radioactive tracers) and percolation tests, potentially above, at and below the repository horizon. The field experiments are complemented by laboratory tests, including, thermal, mechanical, hydrological and rock water interaction.

Observations of the natural system and site data collected since 1978 suggest that the natural system is robust, and that numerical models and calculations will be able to bound many of the uncertainties for radiological safety evaluations, leading to enhanced confidence in the performance of the site. No major unexpected conditions have been encountered; tunneling and testing are confirming the hypotheses on site conditions described in the 1986 Environmental Assessment and the 1988 Site Characterization Plan. Recent performance assessments[10] have led to increased confidence that a Yucca Mountain repository would contain and isolate radioactive waste and would meet a reasonable EPA standard. Disruptive events, such as volcanism or seismicity, are considered unlikely to adversely impact performance. The project recently completed an independent external expert elicitation on probabilistic volcanic hazards[16] that affirmed the project position on low (order of 10^{-8} per year) probability of a volcanic event penetrating the repository. An improved site and engineering data base supports performance assessment calculations that provide more realistic bounding conditions.

Site hydrologic models[17] indicate groundwater flux is likely to be limited and very low at the repository horizon, possibly with local exceptions. The results of modeling and field investigations support a conclusion that infiltration could be diverted laterally away from the repository horizon, owing to the distinctly different hydrologic properties at the contact between coarse-grained and permeable non-welded tuff, and underlying and overlying fine-grained and relatively impermeable welded tuff layers (Fig. 27.3). These hydrologic conditions have the potential to buffer the effects of increased infiltration that potentially could occur as a result of climatic changes. Exploratory studies facility tunneling progress has greatly increased the opportunity for underground observations and confirms constructability and geologic characteristics with no observable water seepage. Also, extensive underground drifting, which provides a greater opportunity for observation and sampling, has supplanted the originally proposed drilling program.

27.5 PLANNED FUTURE WORK

Program scientists and management today believe that a reduction in the scope of the characterization program from that originally proposed is justifiable and desirable. This reduced scope is supported both through increased understanding gained through progress in the characterization program and through realignment of licensing expectations with the information that can reasonably be obtained at different phases of the program[18]. The current understanding of the importance of the individual elements of the disposal system is better and more quantitative than at the time the Site Characterization Plan was written. Some of the uncertainties identified in the Site Characterization Plan are now known to be not as important as others. Performance assessments and modeling have identified the most important uncertainties and methods to bound resolution of other uncertainties without executing every activity of the extensive characterization program originally described in the Site Characterization Plan. What has been learned leaves fewer, but still technically important, questions to be answered about significant features and processes of the natural geologic, hydrologic, and engineered components that would be part of a potential Yucca Mountain repository.

In addition to new site characterization information and performance based analyses capable of evaluating total system performance, other developments such as updated repository and waste package conceptual designs, and considerations related to the change from a release standard to a dose or risk-based standard, with an as-yet unspecified regulatory time frame, have contributed to the need to refine the strategy for evaluating waste containment and isolation.

The updated strategy currently being developed maintains a number of fundamentals of the original strategy. The updated strategy continues to recognize the important role of the relatively dry conditions at Yucca Mountain, which contributed to the site originally being selected for characterization studies. The updated strategy also continues to recognize the geochemical setting provided by Yucca Mountain as important to determining the rate at which radionuclides may be released into the environment in the future, when containment by the waste packages is eventually lost.

The current program is structured around a series of major products leading to an eventual license application. One of the most visible major products in the near term is a Viability Assessment to be completed in 1998. The Viability Assessment has four components:

1. Design of critical parts of the repository, waste packages, and engineered barrier system;
2. Performance assessment that incorporates current knowledge of natural features, processes, and responses at Yucca Mountain, and that evaluates the long term performance of the total natural and engineered system;
3. Total System Life Cycle Cost estimate for the construction, operation, and closure of the repository; and
4. Licensing Plan to lay out tests, design activities, or other actions needed to complete an initial license application for submittal to the United States Nuclear Regulatory Commission.

The purpose of the Viability Assessment is to provide policy makers with an integrated view of a repository system, its estimated performance capabilities, and the associated cost and schedule. If policy makers accept and endorse that assessment, it is anticipated that resources will be committed to ensure continued progress toward the license application.

Current project planning also reflects the need to complete field studies and analyses to reduce uncertainty and enhance understanding of system performance to support the assessment of system safety needed for the license application. The larger technical questions identified in recent total system performance assessments[10] as key to evaluating repository and waste package performance are related to the following attributes of the system:

1. Rate of water seepage into the repository;
2. Integrity of waste packages (containment);
3. Rate of release of radionuclides from waste in the breached waste packages;
4. Radionuclide transport through engineered barriers and natural barriers; and
5. Dilution in the groundwater below the repository.

The refined waste isolation and containment strategy will also address what approach will be taken to gathering data and developing models to make better predictions of these attributes over time. As the repository generates heat and then gradually returns to ambient temperatures, it is expected that at least the first four of these attributes will be affected, changing their relative importance to system performance as a function of time.

In the absence of a definitive compliance standard for geologic disposal, the United States Department of Energy is defining waste containment and isolation for purposes of conducting the viability assessment in such a way as to be independent of the specific compliance measures that eventually will be promulgated[19]. Waste containment has been defined as: the near-complete containment of radionuclides by waste packages for several thousands of years. Isolation has been defined with a system-level safety goal as: an acceptable dose to a member of the public living near the site. Quantitative dose modeling results will be used to evaluate compliance with applicable standards; more attention will be paid to evaluating potential doses for the first ten thousand years. However, calculations will be carried out over longer times in order to provide qualitative insight into peak dose potential, and to support system enhancement studies.

The United States Department of Energy recognizes there are issues which cannot be completely resolved in the 1998 Viability Assessment. Scientific and engineering studies will continue to be conducted to guide confirmation of or revision to the basis for modeling performance of the repository system. There is an expectation that additional data and analyses will be required to support a license application. It also recognizes that, if a license is granted, confirmatory technical work will continue beyond the time of license application into the construction and operational phases of the repository.

27.6 CONCLUSIONS

Following amendment in 1987 of the legislation authorizing characterization of sites for a repository, the United States' high-level waste program focused on Yucca Mountain in the southwestern United States as the single site under consideration. The attributes of Yucca Mountain that made it technically attractive nearly twenty years ago continue to be the technical underpinnings of the strategy for long term waste containment and isolation. Significant progress has been made in the characterization of Yucca Mountain as a potential site for a mined geologic repository. Conditions encountered in the exploratory studies facility tunnel at repository depth are consistent with expectations of such a facility constructed in the unsaturated zone. Total system performance assessments of the long term behavior of a repository at Yucca Mountain continue to mature, and have provided significant guidance in helping define priorities in the test programs and design solutions for the engineered barriers.

The technical strengths of the Yucca Mountain site depend on limited water available to contact the wastes and a corresponding high potential for isolation of the wastes. Today, the United States regulatory approach to long term compliance is uncertain. While the United States Nuclear Regulatory Commission regulations are in place, the United States Environmental Protection Agency standards for disposal safety are remanded. Actions underway to develop a new standard for disposal safety are reopening issues fundamental to the structure of the regulatory approach.

The United States high-level waste program regulations were, in the past, based on a relatively long time frame of regulatory interest, 10,000 years, and assessed compliance against limits on total system releases at an accessible environment, located five kilometers from the repository. The National Academy of Sciences[9] recommendation that the United States adopt a dose based standard for postclosure compliance for a repository has raised issues relative to the regulatory time frame, dose and risk, the definition of the reference biosphere, human intrusion and the quantitative treatment of natural processes and events. Deliberation of these issues is expected to be intense and time consuming, and fundamental re-evaluation of the United States approach to long term compliance should not be unexpected.

Technically, the Yucca Mountain site remains attractive because of its great potential to isolate wastes. However, there are significant concerns about the ability to bring to closure a regulatory proceeding that could have to deal with what are unprecedented time frames in the context of regulation. The potential for a geologic disposal standard that could introduce a need to rely on dilution in a closed hydrologic basin to meet a dose based standard takes the United States high-level waste program full circle back to the promulgation of the United States Environmental Protection Agency standards for geologic disposal. In those proceedings, a dose based standard was considered to be an inappropriate policy that could increase overall population exposures by encouraging disposal methods that would enhance dilution of any radionuclides released[5].

The extent to which the Yucca Mountain site eventually can be shown to be in compliance with a regulation that is evolving amid questions about the very nature of the regulatory structure that has been the basis for selection of the site, and assessment of its performance for nearly twenty years, is a significant concern. This reassessment is occurring even as the geologic and engineering disciplines are beginning to evolve data sets that are unprecedented in depth, breadth and specificity for evaluating the Yucca Mountain site for its waste isolation potential.

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